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May 74

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EPA-520/3-75-003

PRELIMINARY ENVIRONMENTAL ANALYSIS OF A GENERIC FUEL REPROCESSING FACILITY

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PRELIMINARY ENVIRONMENTAL ANALYSIS OF A GENERIC FUEL REPROCESSING FACILITY

R. Cooperstein, R. C. Erdmann, R. R. Fullwood
SAI Services
May 1974

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FOREWORD

The Office of Radiation Programs carries out a national program designed to evaluate the exposure of man to ionizing and nonionizing radiation, and to promote the development of controls necessary to protect the public health and safety and assure environmental quality.

Office of Radiation Programs technical reports allow comprehensive and rapid publishing of the results of intramural and contract projects. The reports are distributed to groups who have known interests in this type of information such as the Nuclear Regulatory Commission, the Energy Research & Development Administration, and State radiation control agencies. These reports are also provided to the National Technical Information Service in order that they may be readily available to the scientific community and to the public.

Comments on this analysis as well as any new information would be welcomed; they may be sent to the Director, Technology Assessment Division (AW-559), Office of Radiation Programs, U.S. Environmental Protection Agency, Washington, D.C. 20460.

W. D. Rowe, Ph.D.
Deputy Assistant Administrator
for Radiation Programs

PREFACE

Nuclear power has become a principal option to satisfy the national need for a clean, safe and reliable energy supply. As a result, the generation of light-water-cooled nuclear power reactors, using enriched uranium fuel, is experiencing rapid growth. This increase in nuclear power reactors will require similar growth in the associated aspects of the fuel cycle such as mining and milling of uranium ore, production of nuclear fuel material, manufacture of fuel elements, shipping, reprocessing of spent fuel elements and waste disposal activities. To date, the controlled and accidental releases of relatively small amounts of radioactivity from operating nuclear power and reprocessing plants have been maintained well below specified limits. However, these operations may impinge to a greater extent on the environment as a result of their anticipated growth.

Projections of the Civilian Nuclear Power Program indicate that the nuclear economy will expand to about 153 gigawatts by 1980 and to about 735 gigawatts by the year 2000. Economic analyses by the AEC and by commercial investors have concluded that generation of electric power by nuclear plants requires reprocessing of spent fuel to recover residual fissile materials for re-use in new fuel elements. Approximate total fuel reprocessing rates in metric tons per year could reach 3500 in 1980 and increase to 15,000 in the year 2000. The total radioactivity due to beta emitters in the accumulated wastes will increase from 210 megacuries (1970) to 18,800 megacuries in 1980 and to 209,000 megacuries in 2000.

The number of reprocessing plants that will be required is a function of the individual plant designs assumed and the amount of spent fuel to be processed. It is assumed, for this study, that one aqueous separations plant will be required for each 1500 metric tons of LWR fuel. A plant of this size should be capable of processing fuel from about 50 power reactors since each typical 1000 MWe LWR will discharge approximately 30 metric tons of fuel each year.

Reprocessing involves destroying the integrity of the spent fuel elements to permit dissolution and separation of the fuel from its metal cladding prior to chemical separation of the useful fissile materials from waste products by some adaptation of the Purex solvent extraction process. Destruction of the integrity of the fuel elements which had been maintained through the cycle in the reactor represents the main source of radioactivity from the nuclear power industry which could potentially enter the environment.

The Environmental Protection Agency, whose charter is to assure protection of the environment by systematic abatement and control of pollution, sponsored a program through its Office of Radiation Programs to perform an environmental analysis study of a generic fuel reprocessing facility in order to project what effects accidents, in such a facility, of potential environmental risk significance, may have on the public health and safety.

This report discusses the probabilities and consequences of hypothetical but credible accidents that could occur in the operation of a generic LWR fuel reprocessing plant which could have potential environmental impacts.

In preparing this report technical data was obtained from numerous sources, nevertheless as might be expected for an analysis of this type, "hard data" were not available in most cases and the authors were required to assume "best judgment values." The limitations which this type of approach places on these data and analysis should be recognized. However, it is our feeling that the methodology and approach used in this report are applicable to environmental analyses at fuel reprocessing plants, and that the information presented despite its limitations is the best available at this time.

SUMMARY

A generic reprocessing plant for light-water-reactor spent fuel utilizing the Purex process has been synthesized from a review of existing plants and those under construction. This model is used to develop a quantitative description of the probability of occurrence of a spectrum of potentially credible accidents and resultant radioactive releases during its operation. The results can be used to determine the potential impact on the environment.

It is concluded from evaluation of the generic design, systems and components that the most probable off-site release pathway that could endanger the public health and safety would be via airborne releases through the ventilation system in the event of accidents.

The postulated credible accidents considered include explosions during various unit operations involving different sources of radioactivity, fires, criticality, loss of cooling to the high level liquid waste facility and accidents which could derive from the occurrence of natural events such as earthquakes, tornadoes and floods.

Accidents which might occur during normal operation were emphasized over those that might occur during shutdown or neptunium batch processing conditions. In addition, no releases during decommissioning or start-up operations were analyzed.

If, during an accident large releases of noble gases or tritium were noted, these were not assessed because they

would normally be released in reprocessing spent fuel. The installation of noble gas collectors, to minimize krypton releases for example, were examined briefly. While such systems exist in the laboratory, production-scale systems are presently not available. Therefore the risk tradeoff between continual release of such gases and possible accidental release of stored quantities of such gases, after some period of radioactive decay, was not conducted.

Hypothetical accident probabilities are estimated by fault tree analysis of the model plant's safety and confinement systems during operation. The expected responses to hypothesized operating transients and postulated accidents, including natural events, are evaluated. Realistic assumptions based upon existing process technology and experience are used in the evaluations to determine the consequent radioactive releases.

If a processing modification is incorporated into the reprocessing cycle it will be a relatively simple matter to estimate the likelihood of releases from such a change for comparison with those documented herein. Thus, a measure of the level of acceptability of a processing change from a release or safety viewpoint is evident from this work.

In such a generic analysis, it is not possible to make a detailed account of the operational and design data which would be pertinent to an in-depth safety analysis. We have attempted, where possible, to utilize existing failure data supplementing this with estimates of how such an incident during processing might take place. Our considerations included types of instrumentation used, human behavior and design. Thus, our results may be suspect in that errors in design, etc., may have been overlooked. Barring these anomalies, the accuracy of our results may vary within a factor of 10 or less from the true situation.

The ventilation system (scrubbers, HEPA filters) played a very prominent role in decreasing the consequences of many of the accidents releasing nonvolatile species. If such releases are determined to be excessive, one should consider installation of additional release mitigating equipment as a positive means of removing hazardous airborne substances.

Since the spent fuel to be reprocessed has substantially less decay heat compared to when it is housed in a reactor, the concerns of decay heat removal at a reprocessing plant, while real are not severe. Interruptions of power for effecting the latter, can be more readily tolerated. Even so, auxiliary power generation equipment is available to provide plant power during emergencies.

Based upon fault tree analysis and consequence calculations, consequence/likelihood plots are drawn for selected isotopes including ruthenium, iodine, plutonium, and other representative actinides and fission products.

By selecting the dominant accidents from these plots and by applying a simplified meteorological case, the dose in rems as a function of distance for a number of pertinent isotopes are plotted utilizing likelihood as a parameter. These data permit quantification of accident impact on the environment for a generic reprocessing plant.

It is recognized by EPA that this report presents only an initial analysis of the potential environmental impact of accidents at a generic LWR fuel reprocessing facility. There remain further analyses which could be completed, using this technique, for both a generic facility and for any specific facility design sited at a particular location. Extensions of the current study to derive population dose estimates and predictions of the health effects which could result from these exposures are examples of two such possible additional studies.

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I. INTRODUCTION

As of June 1, 1973, there were 33 operating reactors, 56 being built and 80 additional reactors on order. Based on nuclear power projections of 300,000 MWe by 1985, additional fuel reprocessing capability is needed to provide for the recovery of fissile material remaining in spent fuel elements[1]. The two existing fuel reprocessing plants, Nuclear Fuel Services and the Midwest Fuel Recovery Plant, have combined capacities for reprocessing the spent fuel discharges from the equivalent of about 35 1000 MWe light water cooled power reactors (1050 MTU/year). The Barnwell Nuclear Fuel Plant, under construction, will have a capacity for reprocessing an additional 1500 MTU/year of spent LWR fuel bringing the total annual reprocessing capacity to only 2550 MTU/year. This will be insufficient capacity for the industry within the balance of this decade thereby requiring the construction of added capability to ensure smooth, economical and timely operation of the nuclear fuel cycle for the anticipated energy requirements.

In the operation of a nuclear power reactor, the buildup of fission products and the depletion of fissile material (U-235 and Pu) requires that for maximum utilization of the fuel, the reactor operator must periodically replace about one-third of the fuel elements and redistribute the remaining partially spent fuel elements throughout the reactor core. The discharged spent fuel elements still contain between one-third and one-fourth of the U-235 in the fuel prior to irradiation and part of the fissile Pu that was produced from U-238. The fuel reprocessing plant recovers the unused fissile material so that it can be recycled in reactor reload fuel. Reprocessing also permits separation and removal of the fission products from the fissile material for conversion into an acceptable form for long term isolation from the biosphere.

The sequential process of transforming uranium ore into usable fuel for nuclear power reactors and the operations to recover unused values of uranium as well as the plutonium and other desirable isotopes produced during irradiation in the reactors constitute the "nuclear fuel cycle"[2]. These operations are generally performed at separate installations in various parts of the country, depending for the most part upon the economics of transportation. The specific components comprising the LWR supporting fuel cycle are shown in Figure I-1, page 3, and include the following:

a) Mining uranium ore

b) Milling and refining ore to produce uranium concentrates (U_3O_8)

c) Production of uranium hexafluoride (UF_6) from uranium concentrates to provide feed for isotopic enrichment

d) Isotopic enrichment of uranium hexafluoride to attain reactor enrichment requirements using the gaseous diffusion process

e) Fabrication of nuclear reactor fuel including: converting UF_6 to uranium dioxide (UO_2), pelletizing, encapsulating in rods and assembling fuel elements

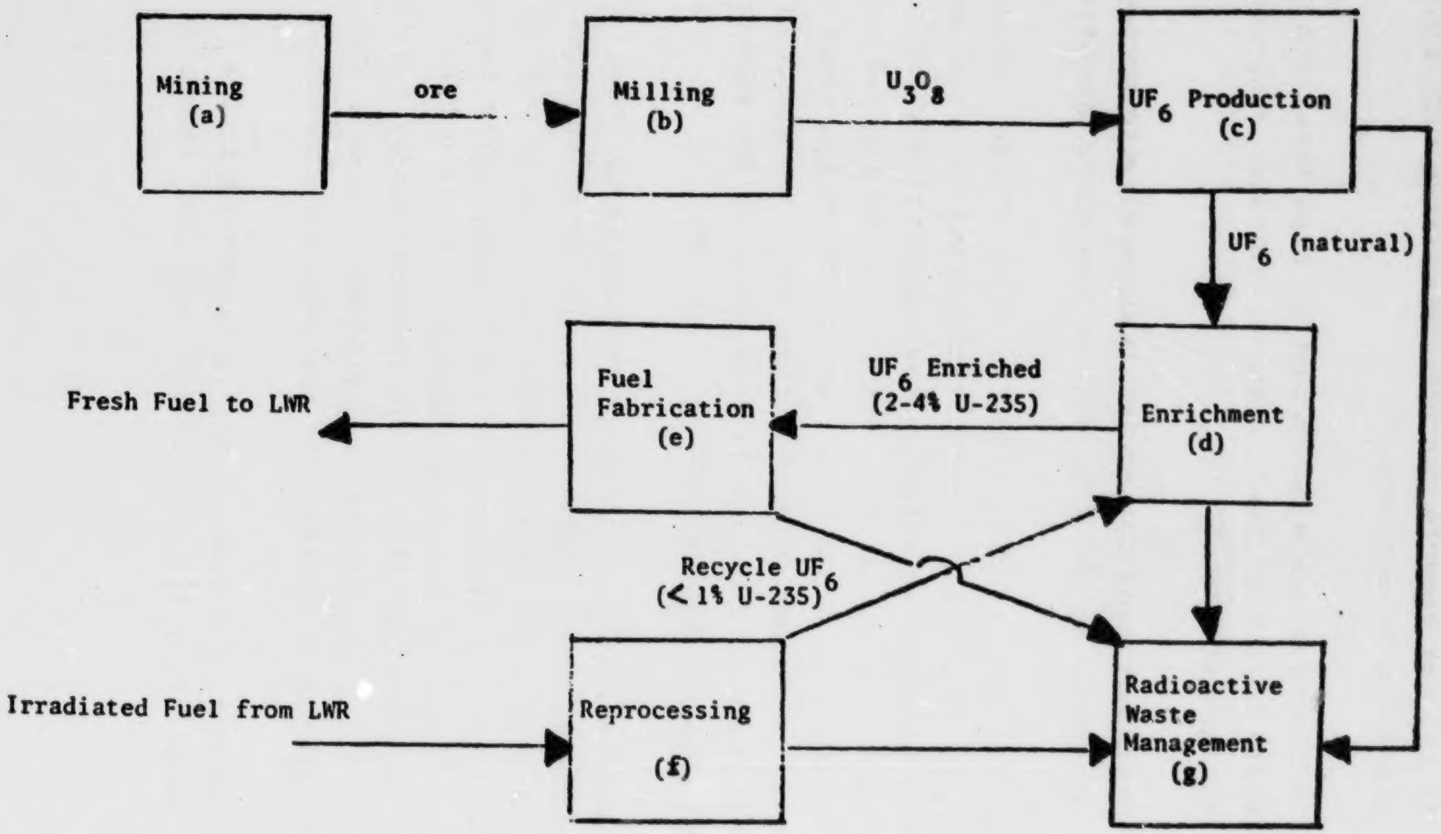

Mining
(a)
ore
Milling
(b)
U_3O_8
UF_6 Production
(c)
UF_6 (natural)
Enrichment
(d)
UF_6 Enriched
(2-4% U-235)
Fuel
Fabrication
(e)
Fresh Fuel to LWR
Recycle UF_6
(< 1% U-235)
Irradiated Fuel from LWR
Reprocessing
(f)
Radioactive
Waste
Management
(g)

FIGURE I-1

Nuclear Fuel Cycle - Light Water Reactors

Uranium Dioxide Fueled - No Plutonium Recycle

f) Reprocessing irradiated fuel and converting uranium to UF_6 for recycle through the gaseous diffusion plant for re-enrichment

g) Radioactive waste management of high level and other than high level wastes, including long-term storage of wastes

h) Transportation activities associated with moving materials to and from each of the above operations.

Uranium milling and refining (beneficiation of raw uranium ore into U_3O_8) is usually done near the mines to avoid the cost penalty involved in shipping the comparatively low value ore over long distances. Uranium ore is refined into U_3O_8 for shipment in drums to a "conversion" plant where it is converted into uranium hexafluoride (UF_6). Uranium in the UF_6 gaseous state is required as feed material for subsequent "enrichment" in the gaseous diffusion process.

Uranium hexafluoride [illegible], although refined as to total uranium content, still contains less than 1% of the fissionable isotope U-235 after "conversion". To be suitable for fabrication into fuel elements for modern power reactors, it must be "enriched" to approximately 2% to 5% U-235 (depending upon the specific design requirements of individual reactors). Thus, UF_6 is fed into enrichment plants where the U-235 isotope is upgraded to the required content.

After enrichment, the gaseous UF_6 is converted into a metallic oxide (UO_2) for fabrication into fuel elements which, together with control rods, structure and moderating components, form the nuclear reactor core.

In the reactor, fuel elements initiate and sustain the controlled fissioning "chain reaction" which produces vast quantities of heat necessary to generate electricity via the steam-powered turbine generators. On the average, one pound of slightly enriched uranium produces approximately 3.2 million kilowatt hours (KWH) of electricity. This compares with about 2.2 million pounds of coal required to generate the equivalent electricity.

During irradiation in the reactor, various fission products are created which tend to lower reactivity over time through absorption of neutrons. Excessive accumulation of fission products and burn-up of U-235 impair the chain reaction and eventually shut down the reactor. Irradiated fuel, therefore, must be replaced periodically after being only partially consumed. As the fuel is irradiated in the reactor core, part of the fertile uranium isotope U-238 is converted to plutonium, a portion of which undergoes fission thereby contributing to the reactor's heat output. The remaining plutonium and other fission products stay intact and become potential byproducts and waste residues.

Economic considerations favor the recovery of the fissile material remaining in the spent fuel elements. The net value of this residual fissile material, after allowing for the costs associated with reprocessing, waste disposal and related transportation services, amounts to about $50,000 per metric ton of irradiated fuel. Thus, during each year a 1500 MTU capacity reprocessing plant is operated at full capacity, it will reclaim fissile material having a net worth ~~of~~ about $75,000,000. Moreover, by recovery of fissile material, such a plant will conserve natural resources equivalent to about a million and one-half tons of uranium ore each year[3].

Reprocessing accomplishes the objectives of:

a. reclaiming the unused uranium and plutonium for subsequent recycling into replacement fuel

b. extracting valuable isotopes such as neptunium and separating the waste fission products from the reusable fissile material

c. concentrating the fission products and associated irradiated wastes to permit safer, less complicated handling for permanent storage and more economical storage that will result in a minimal impact on the environment.

This study is concerned with determining the probability of accidents associated with a generic light-water reactor irradiated fuel reprocessing plant and the consequences of these accidents to the environment. This analysis will permit a quantitative risk comparison with other parts of the nuclear fuel cycle and to other risks accepted by society. The study is limited to developing the accident risk envelope for a generic reprocessing plant that could impact on the environment. Fault tree analyses are used for the accident probability predictions. Only those trees needed in the development of the risk envelope are evaluated although a complete set of trees is presented to serve as an aid for future work in risk assessments of fuel reprocessing facilities.

II. DESCRIPTION OF THE GENERIC FUEL REPROCESSING PLANT AND CONTROLS

Public health and safety is a principal concern in reprocessing spent irradiated fuel to a similar extent as for nuclear power reactors. In fact, the reprocessing plant may be a greater source of radioactivity in effluents than a reactor[4]. Thus, adequate safety margins are included in the design of reprocessing plants to prevent accidents and to assure that acceptable protection systems will function reliably to mitigate the consequences of accidents, if they should occur, because of multiple system failures or noncompliance with procedures provided to prevent accidents[5-7].

The function of a fuel reprocessing plant is to recover the residual fuel materials, uranium and plutonium, in a form suitable for reuse and to isolate radioactive wastes for storage and ultimate permanent disposal. The basic elements of reprocessing are illustrated in Figure II-1, page 8. This simplified flow diagram is generally applicable to any of the process techniques which have been applied for spent fuel reprocessing. The generic reprocessing technique for this study is the Purex process, a solvent extraction process, which has been in use for two decades in this country and is in use in other countries where spent fuels are processed[8].

1. Plant Description

The following assumptions are made regarding the generic plant:

- the facility would be sited in conformance with AEC siting criteria as expressed in Part 100, Title 10 of the U.S. Code of Federal Regulations (10 CFR 100)[9].
- sufficient water supply would be available for plant operation

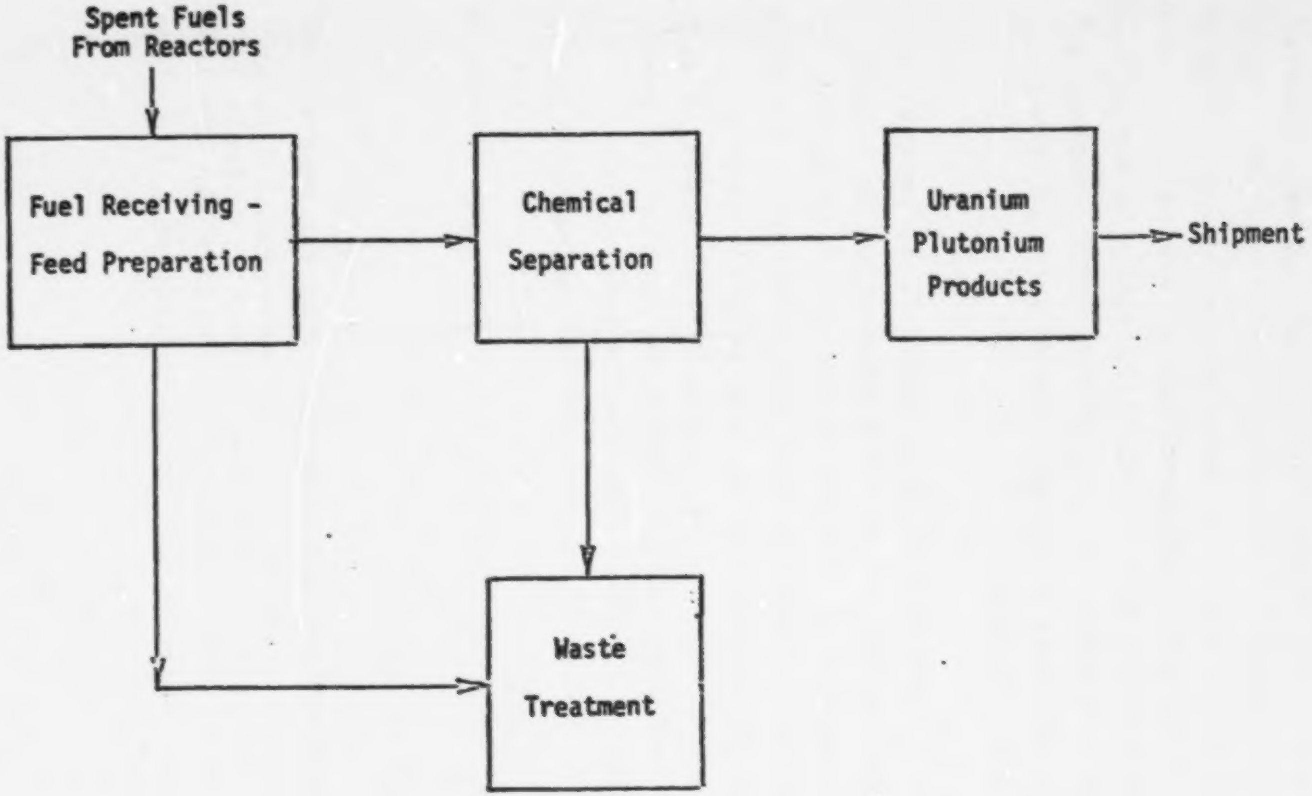

Spent Fuels
From Reactors
Fuel Receiving -
Feed Preparation
Chemical
Separation
Uranium
Plutonium
Products
Shipment
Waste
Treatment

FIGURE II-1

Fuel Reprocessing - Generic Flow Diagram

- it would have a processing rate of 5 metric tons of heavy metal (uranium and plutonium) per day of spent fuel from light-water power reactors
- the major facilities on the site would be:
  1. fuel receiving and storage facility
  2. main processing building housing the reprocessing, product storage and waste solidification equipment
  3. radioactive-area-ventilation-air filtration and discharge system
  4. high level radioactive liquid waste storage
  5. offices
  6. warehouse and shops
  7. steam-generating plant
  8. cooling towers
  9. a retention basin
  10. product conversion facilities

All processing equipment and systems for processing irradiated fuel elements, except for waste tank facilities, will be housed in the process, fuel receiving, and storage station building by the nature of the radiochemical operation. Because of the radioactivity that must be handled, many of the process systems must be heavily shielded; the equipment must be operated, and some of it must be maintained from stations that are remote from the equipment itself. Thus, as illustrated in Figure II-2, page 10, the facility is likely to be a massive concrete structure. This particular building was the proposed Atlantic Richfield Reprocessing Center which has the basic features of the Hanford and Barnwell Nuclear Fuel Plants[10].

The model plant for this study is synthesized from the features of the three plants scheduled to be operational by the mid-1970's. These are the Barnwell Nuclear Fuel Plant (BNFP), the Nuclear Fuel Services Plant (NFS) and the Midwest Fuel Recovery Plant (MFRP). Their respective reprocessing capacities are 1500, 750 and 300 metric tons per year of low-enrichment uranium fuels irradiated in light-water power reactors[5-7]. The assumed model most closely resembles the BNFP because

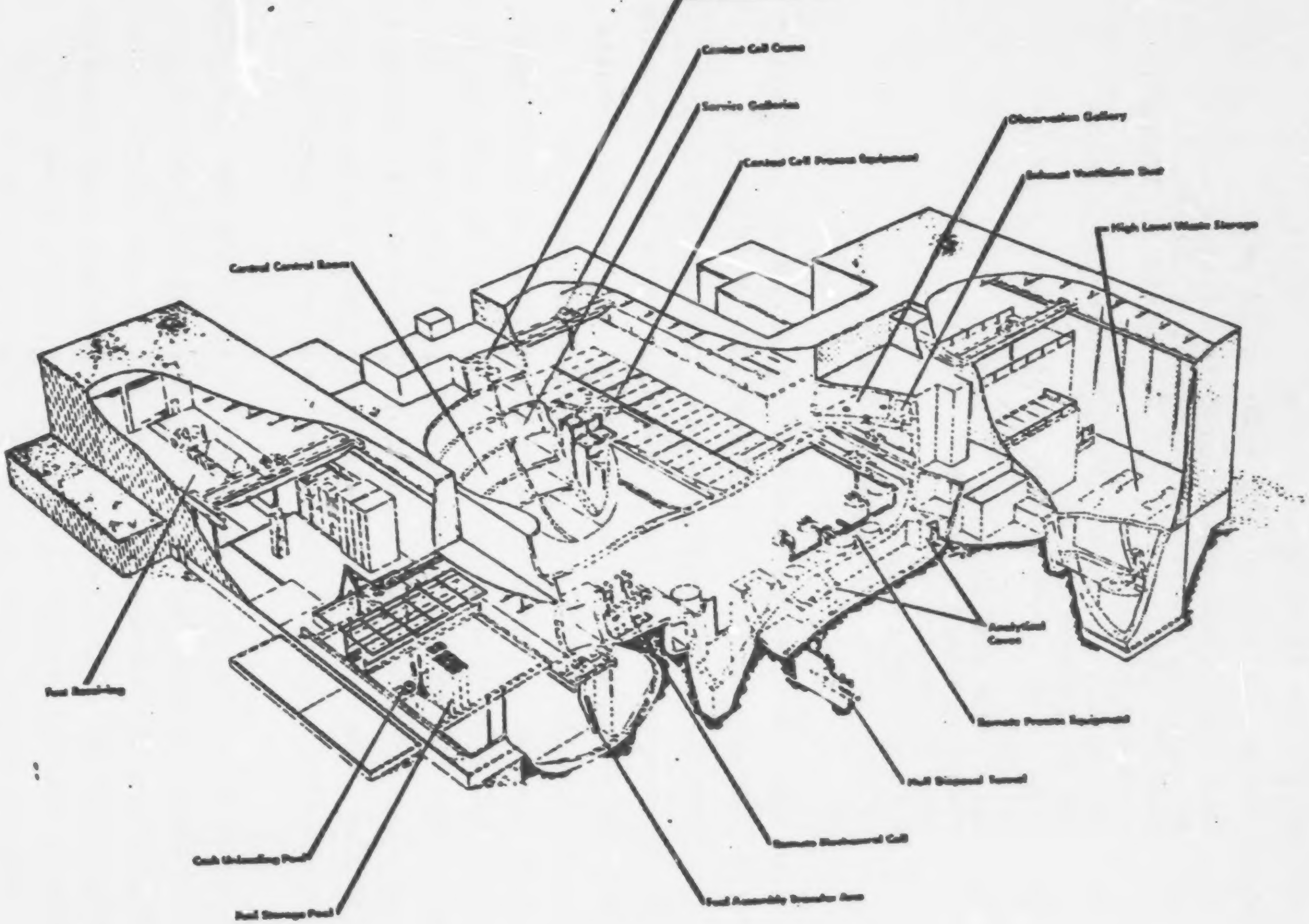

Cold Chemical Preparation
Canyon Cell Crane
Service Galleries
Canyon Cell Process Equipment
Observation Gallery
Exhaust Ventilation Duct
High Level Waste Storage
Central Control Room
Fuel Building
Analytical Caves
Remote Process Equipment
Hull Disposal Tunnel
Remote Mechanical Cell
Fuel Assembly Transfer Area
Cask Unloading Pool
Fuel Storage Pool

FIGURE II-2
ARC PROCESS BUILDING

of its larger capacity, which is used for the purpose of estimating the fractional releases of radioactive materials for the accidents considered. This choice is based on the belief that future plants will be designed for such a capacity to effect low unit processing costs. High capacity, small volume equipment will be used to minimize plant inventory of both reactor fuel and process reagents thereby ensuring a greater degree of overall safety and economy. Future plants may have different characteristics from those used in this study; however, it is expected that the deriv[illegible] quantities of radionuclides that could be released in potential credible accidents will remain unchanged or decrease as a result of advancing technology[11].

2. Comparison of Potential Hazards in a Reprocessing Plant and a Nuclear Power Plant

Performance criteria for engineered safety in reprocessing plants are based upon those proposed for nuclear reactors (Appendix A, 10 CFR 50; Code of Federal Regulations[9]) although the function and design of reprocessing plants are significantly different. Potential hazards in a separations facility differ considerably from those anticipated in power reactors due to the specific conditions found in the reprocessing plant. Examples of these differences follow:

- The potential for a nuclear criticality is very low in a reprocessing plant albeit fissionable isotopes are present in quantity and are separated and purified in the course of operations. The use of soluble and fixed poisons, favorable geometry, concentration control and mass control mitigates the possibility of nuclear criticality where a nuclear chain reaction could take place.

- Fuel reprocessing operations are generally of such a nature that the rate of approach of critical parameters (concentration, temperature, acidity, etc.) to prescribed upper limits will be relatively slow compared to power reactors. Although reliable instrumentation and control provisions are required, very rapid response is not necessary. The effect of exceeding prescribed operating limits does not usually present an immediate hazard.

- The cladding, which serves as the primary barrier to fission product escape from fuel in a reactor, must be breached in the reprocessing operation to recover the fissile materials. The potential hazard from having mobile radioactive materials in plant process systems, however, is relatively low since the systems do not contain large amounts of stored energy (like power reactor primary coolant systems) which could provide the driving force of radioactivity dispersal.

- In some instances, plant process streams will be corrosive. System failure due to corrosion with subsequent radionuclide releases may not be severe for the streams are doubly contained for leak tightness and operate at low pressures.

- Flammable and/or chemically reactive materials are used in fuel reprocessing operations. Well developed technologies, however, are used for assuring that potential hazards from use of such materials are appropriately controlled.

- The potential hazards from loss of plant cooling capability are low due to the lower stored energy levels of the limited fissile inventory in process which has already undergone extended decay times. Emergency power supply availability will enhance safety assurance by providing positive off-gas release control and continuity of process condition surveillance and safe shutdown conditions. Continuous cooling is not critical although necessary for high level wastes which are stored on an interim basis. Cooling is used to avoid boiling in the high level liquid wastes and/or overheating of the containment vessels. Overheating could compromise the integrity of the storage tanks resulting in uncontrolled releases of waste fission products to the environment.

- Fuel reprocessing operations are performed at low temperatures - limited to normal boiling points at atmospheric pressure - and at low pressures - limited by subatmospheric pressure maintained in the process system.
- Fission product wastes produced during irradiation of power reactor fuel are separated in the reprocessing plant and large quantities of these materials must be stored and controlled. Process operations are performed to reduce the volumes of solutions of such wastes to more manageable volumes which will not be released to local water courses. Alternatively, the wastes will be converted to low mobility forms (solids) for safe on-site retention until transferred to authorized permanent disposal facilities.
- Off-gas treatment processes and controlled effluent releases are required to assure that gaseous products which are not amenable to immobilization are not released from the plant at rates exceeding prescribed limits.

- The Purex process basically involves solutions and as such makes the recovery of tritium especially difficult.

3. Process Description

The overall process function is to recover the contained uranium and plutonium from spent fuel assemblies. The Purex solvent extraction process will be used to separate the latter from fission products. To accomplish the overall function, several processes must be used. They are described in the following and are illustrated by the simplified block diagram in Figure II-3, page 15, which is the particular flow diagram for the BNFP facility. For comparative purposes, the block diagrams for the MFRP and NFS processes are illustrated in Figures II-4 and II-5, pages 16 and 17. Principal similarities and differences among the commercial plants and the assumed model are listed in Tables II-1 and II-2, pages 18 and 19.

- Fuel Receipt and Storage. Irradiated fuel assemblies arrive at the reprocessing plant in shielded casks. These are monitored for outside contamination, cleaned, removed from the carriers and submerged in a pool of water for unloading the fuel assemblies. The cask is opened, the fuel assemblies removed and placed in storage canisters. The canisters are moved to a fuel storage pool where they are held until the fuel is scheduled for reprocessing.

- Mechanical Disassembly. When scheduled for reprocessing, the fuel assembly is remotely transferred to a mechanical facility where it is sheared, as rods or as fuel bundles, into short lengths to expose the fuel to the dissolvent.

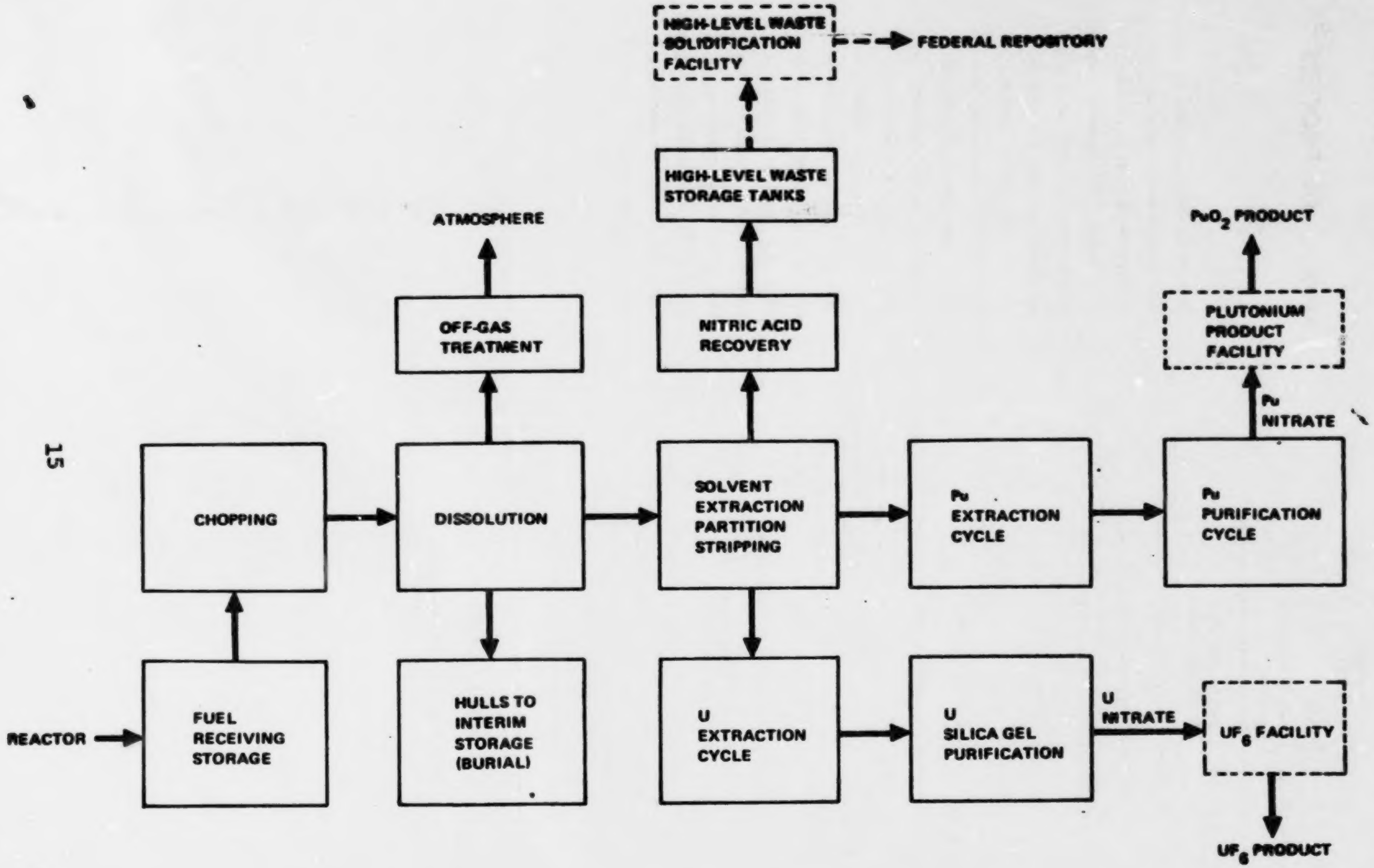
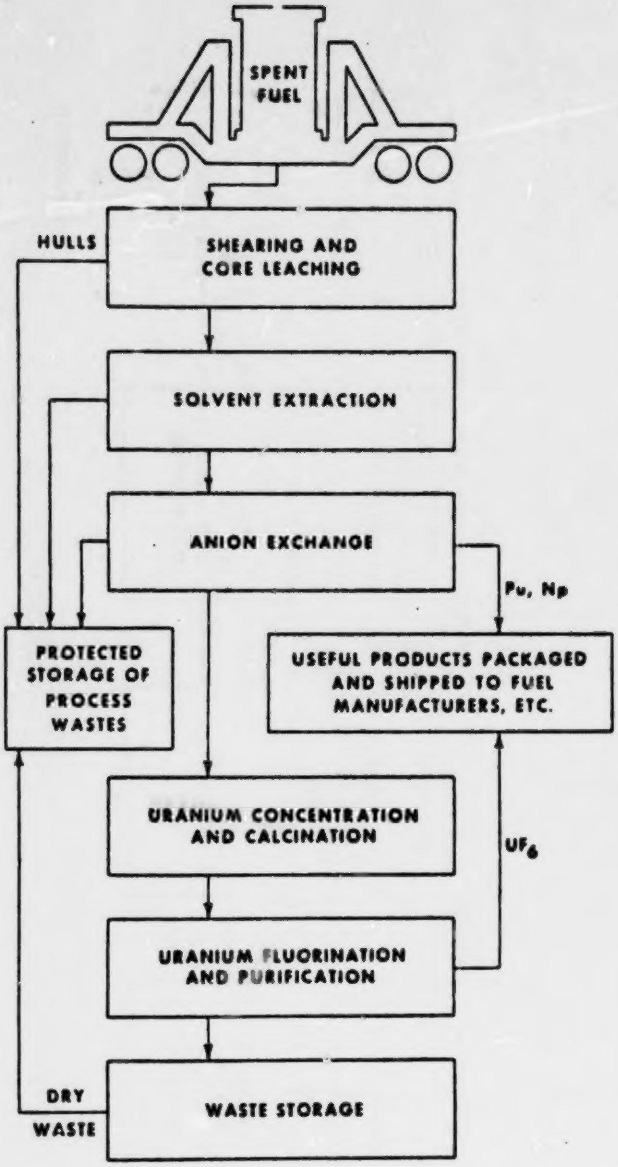

HIGH-LEVEL WASTE SOLIDIFICATION FACILITY
FEDERAL REPOSITORY
HIGH-LEVEL WASTE STORAGE TANKS
ATMOSPHERE
OFF-GAS TREATMENT
NITRIC ACID RECOVERY
PuO2 PRODUCT
PLUTONIUM PRODUCT FACILITY
Pu NITRATE
CHOPPING
DISSOLUTION
SOLVENT EXTRACTION PARTITION STRIPPING
Pu EXTRACTION CYCLE
Pu PURIFICATION CYCLE
REACTOR
FUEL RECEIVING STORAGE
HULLS TO INTERIM STORAGE (BURIAL)
U EXTRACTION CYCLE
U SILICA GEL PURIFICATION
U NITRATE
UF6 FACILITY
UF6 PRODUCT

FIGURE II-3

Simplified Flow Diagram of BNFP Process[7]

FIGURE II-4

MFRP GE PROCESS [6]


SPENT FUEL
HULLS
SHEARING AND CORE LEACHING
SOLVENT EXTRACTION
ANION EXCHANGE
Pu, Np
PROTECTED STORAGE OF PROCESS WASTES
USEFUL PRODUCTS PACKAGED AND SHIPPED TO FUEL MANUFACTURERS, ETC.
URANIUM CONCENTRATION AND CALCINATION
UF$_6$
URANIUM FLUORINATION AND PURIFICATION
DRY WASTE
WASTE STORAGE

General Electric's proposed Aquafluor process makes use of aqueous and fluoride volatility fuel recovery technologies. The process uses well-demonstrated fuel recovery techniques including fuel shearing and leaching, solvent extraction, anion exchange, calcination, fluorination, and uranium hexafluoride distillation. Anion exchange is used in the recovery and purification of plutonium and neptunium. Uranium will be converted to the volatile uranium hexafluoride (UF_6) and purified to make it suitable for toll enrichment. The high level liquid wastes will be converted to solid form and stored temporarily on-site in a waterfilled concrete storage basin.

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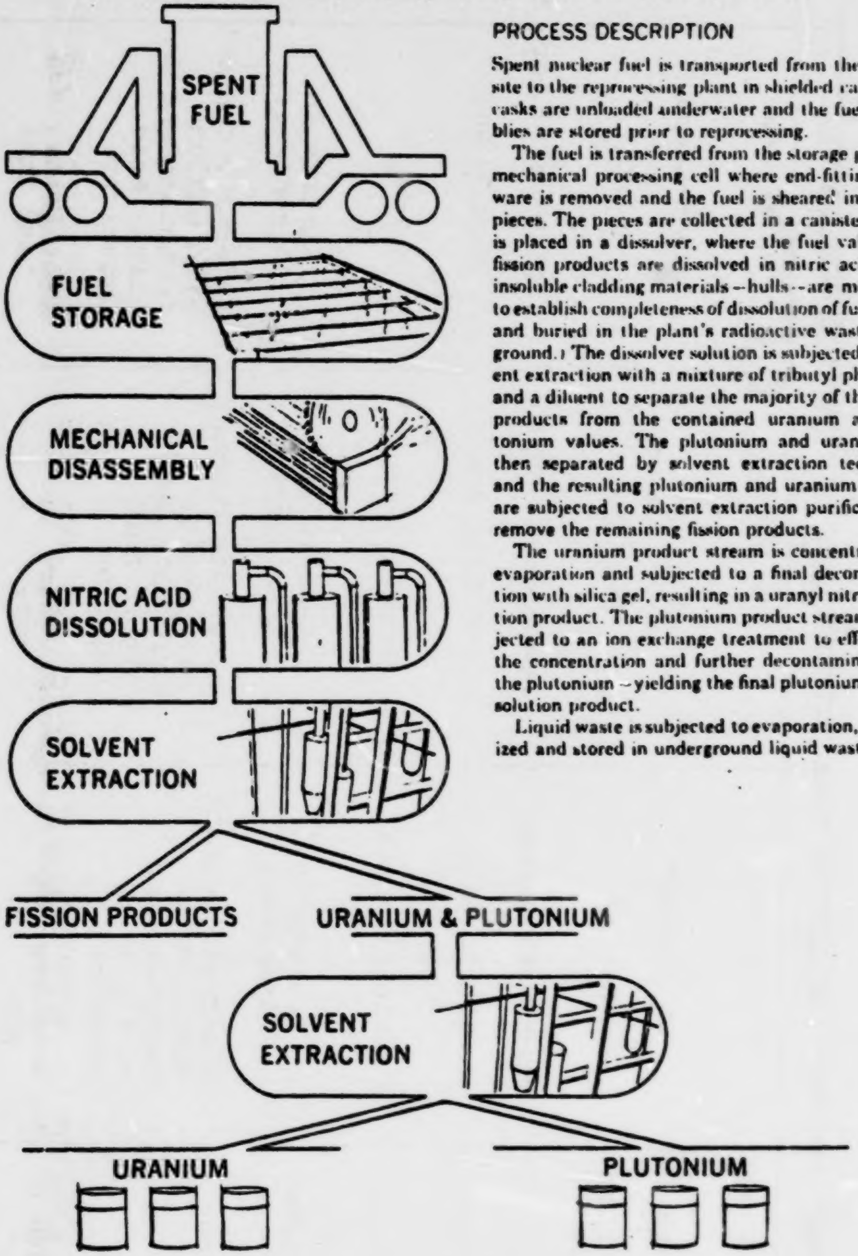
FIGURE II-5

NUCLEAR FUEL SERVICES, INC., SPENT FUEL PROCESS[5]



PROCESS DESCRIPTION

Spent nuclear fuel is transported from the reactor site to the reprocessing plant in shielded casks. The casks are unloaded underwater and the fuel assemblies are stored prior to reprocessing.

The fuel is transferred from the storage pool to a mechanical processing cell where end-fitting hardware is removed and the fuel is sheared into small pieces. The pieces are collected in a canister, which is placed in a dissolver, where the fuel values and fission products are dissolved in nitric acid. (The insoluble cladding materials--hulls--are monitored to establish completeness of dissolution of fuel values and buried in the plant's radioactive waste burial ground.) The dissolver solution is subjected to solvent extraction with a mixture of tributyl phosphate and a diluent to separate the majority of the fission products from the contained uranium and plutonium values. The plutonium and uranium are then separated by solvent extraction techniques and the resulting plutonium and uranium streams are subjected to solvent extraction purification to remove the remaining fission products.

The uranium product stream is concentrated by evaporation and subjected to a final decontamination with silica gel, resulting in a uranyl nitrate solution product. The plutonium product stream is subjected to an ion exchange treatment to effect both the concentration and further decontamination of the plutonium--yielding the final plutonium nitrate solution product.

Liquid waste is subjected to evaporation, neutralized and stored in underground liquid waste tanks.

TABLE II-1

PRINCIPAL SIMILARITIES OF THE MODEL SEPARATIONS FACILITY COMPARED TO OTHER COMMERCIAL REPROCESSING FACILITIES

Comparison Item	Model	BNFP	NFS	MFRP
Fuel Unloading and Storage	Underwater	Underwater	Underwater	Underwater
Headend Process	Chop-Leach	Chop-Leach	Chop-Leach	Chop-Leach
Stored Fuel Criticality Safety	Spacing	Spacing	Spacing	Spacing
Fuel Chopping	Mechanical Shear	Mechanical Shear	Mechanical Shear	Mechanical Shear
Fuel Dissolution Material	Nitric Acid	Nitric Acid	Nitric Acid	Nitric Acid
Fuel Dissolution Technique	Semicontinuous	Semicontinuous	Batch	Semicontinuous
Fuel Dissolution Equipment	Baskets in Dissolvers	Baskets in Dissolvers	Baskets in Dissolvers	(No Similarity) Leaching trough
Solvent Cleanup	Alternate Contact with Sodium Carbonate and Nitric Acid Solutions	Alternate Contact with Sodium Carbonate and Nitric Acid Solutions	Alternate Contact with Sodium Carbonate and Nitric Acid Solutions	Alternate Contact with Sodium Carbonate and Nitric Acid Solutions
Final Exhaust Filters	Roughing and HEPA	Roughing and HEPA	Deep Fiberglass and HEPA	(No Similarity) Sand Filter

TABLE II-2

PRINCIPAL DIFFERENCES OF THE MODEL SEPARATIONS FACILITY COMPARED TO OTHER COMMERCIAL REPROCESSING PLANTS

Comparison Item	Model	BNFP Separations Facility	NFS	MFRP
Location	Compliance with 10 CFR 100	South Carolina	New York	Illinois
Design Capacity	5 MTU/day	5 MTU/day	3 MTU/day	1 MTU/day
Shearing	Entire fuel elements including end fittings	Entire fuel elements including end fittings	End fittings may be removed before shearing	Pins removed from fuel elements before shearing pins only
Criticality Control During Dissolution	Soluble poison	Soluble poison	Geometric limitations	Geometric limitations
Fuel Dissolution Technique	Semicontinuous	Semicontinuous	Batch	Semicontinuous
Fuel Dissolution Equipment	Baskets in Dissolvers	Baskets in Dissolvers	Baskets in Dissolvers	Vibratory Leacher Tray
HA Contactor	Centrifugal Contactor	Centrifugal Contactor	Pulse Column	Pulse Column
Partitioning	Ion Exchange	Electropulse column	Solvent extraction with chemical valence adjustmen	Ion exchange
Interim High-Level Waste Storage Form	Acidic solution (1-5 molar)	Acidic solution (1-5 molar)	Acidic solution	Calcined solid
Iodine Removal from Process Off-gas	Mercuric Nitrate Iodine Scrubbers plus Iodine Silver Zeolite Adsorption Beds	Mercuric Nitrate Iodine Scrubbers plus Iodine Silver Zeolite Adsorption Beds	Mercuric nitrate scrubbers plus silver zeolite adsorption beds.	Sodium Hydroxide Scrubbers. Heated Silver Zeolite
Process Vent Filters	Roughing and HEPA	Roughing and HEPA	Multiple HEPA	Packed Fiberglass Filter
Final Exhaust Filters	Roughing and HEPA	Roughing and HEPA	Deep Bed Fiberglas plus HEPA	Sand Filter
Feed Clarification	Centrifuge	Centrifuge	None	None
Tritium Disposal	As vapor up stack	As vapor up stack	As water to creek	As vapor up stack
Uranium Product Form	Hexafluoride	Nitrate solution	Hexafluoride	Hexafluoride
Liquid Effluent	Noncontaminated	Noncontaminated	Contaminated	Noncontaminated

The fuel segments fall into or are fed into a dissolver vessel.

- Fuel Dissolution. The segmented fuel containing the plutonium, neptunium and fission products formed during irradiation, as well as unspent uranium is dissolved out of the cladding hulls with nitric acid to form the feed for subsequent remote liquid-liquid solvent extraction steps. After dissolution, the undissolved cladding hulls, made of zirconium, zirconium alloys or stainless steel, are separated from the solution, rinsed, monitored for residual fissile material and transferred to a protected interim vault storage area. Gases generated during the dissolutions are channeled to an off-gas treatment system. This system contains decontaminating units such as condensers, scrubbers, chemical traps for iodine removal and particulate filters which remove radioactive gases and particulates other than inert gas, e.g., Kr-85 and tritium, to levels below allowable release limits before being exhausted to the atmosphere. Nitrogen oxides formed during the dissolutions are also retained by the ventilation treatment system to minimize their release to the atmosphere.

- Solvent Extraction. The chemically adjusted aqueous feed solution is then subjected to a Purex-type extraction. It is contacted countercurrently in a centrifugal contactor with an organic solution of tributyl phosphate (TBP) dissolved in normal paraffin hydrocarbon diluent (dodecane). The organic solution preferentially extracts the tetravalent plutonium and hexavalent uranium, leaving about 95% of the fission products in the aqueous solution. The organic solution from the extraction passes through a scrubbing column where it is washed with additional nitric acid solution. This step removes about 96% of the residual extracted fission products from the organic product solution. The wash solution is recycled back to the centrifugal contactor. The aqueous solution leaving the codecontamination cycle contains about 99.8% of the fission products from the initially dissolved solution. It is routed to a high level waste treatment system where it is concentrated for protected interim liquid waste storage and/or calcined to an immobile solid form, loaded into specially designed containers and transferred to protected storage facilities for ultimate authorized off-site disposal. A simplified block diagram illustrating the solvent extraction cycle is shown in Figure II-6, page 22.

- Product Separation. Anion exchange or electrochemical reduction is used to partition plutonium and uranium into separate streams following the codecontamination step. In the former operation, solution from the solvent extraction system concentrator is fed through a cooler to a series of semicontinuous ion exchange contactors, in an ion-exchange cell, where the plutonium is sorbed on the resin as anionic $Pu(NO_3)^{--}$ and subsequently removed as a nitrate solution for concentration and loadout. Alternatively, the organic solution from the codecontamination step can be passed through a partitioning column where tetravalent plutonium is electrochemically reduced to the less extractable trivalent state. The reduced plutonium is then stripped into an aqueous nitric acid solution containing hydrazine as a holding reductant. The organic uranium containing solution is then stripped into acidified water. Electrochemical reduction and solvent extractions eliminate the need for chemical

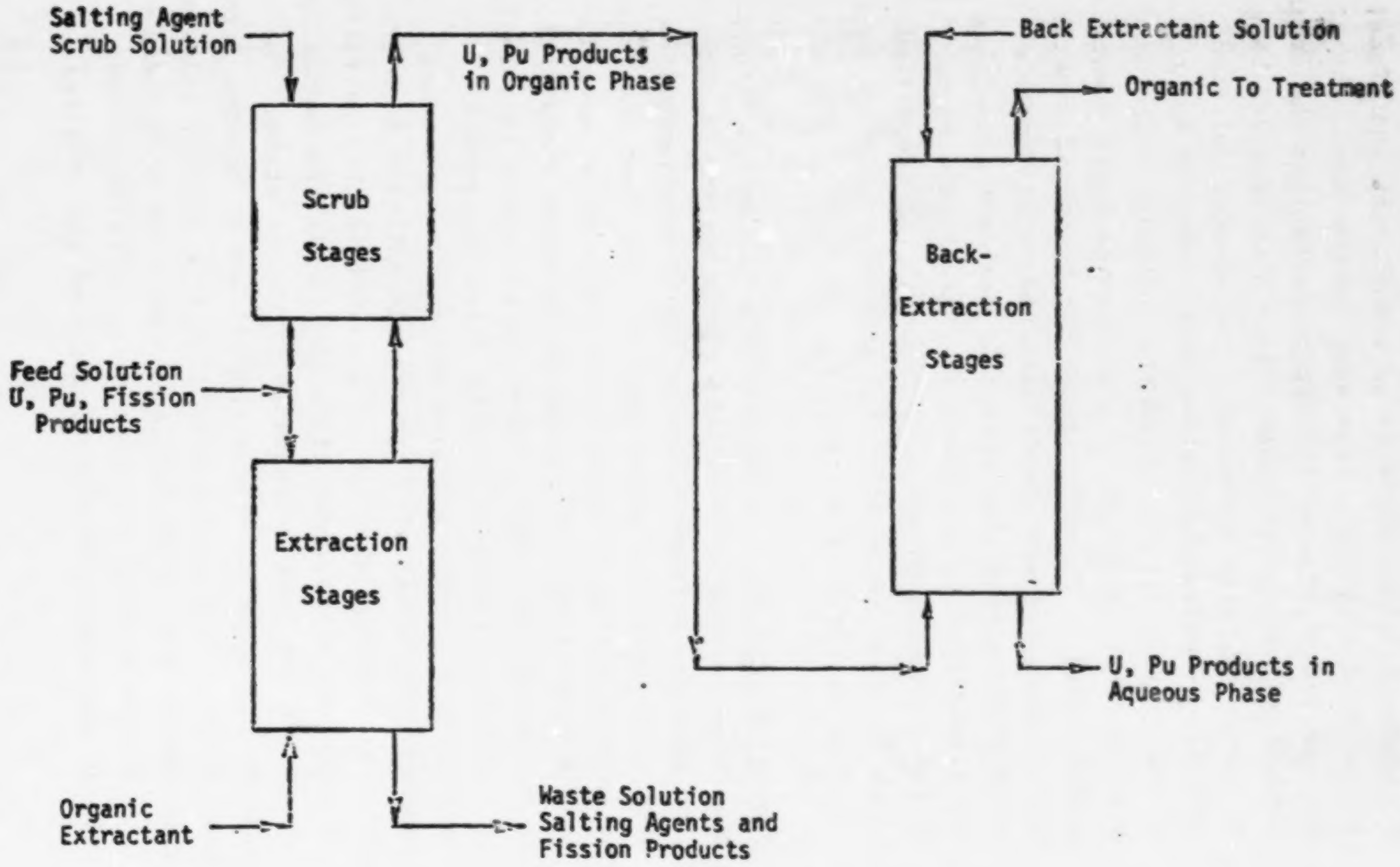

Salting Agent
Scrub Solution
Scrub
Stages
U, Pu Products
in Organic Phase
Back Extractant Solution
Organic To Treatment
Back-
Extraction
Stages
Feed Solution
U, Pu, Fission
Products
Extraction
Stages
U, Pu Products in
Aqueous Phase
Organic
Extractant
Waste Solution
Salting Agents and
Fission Products

FIGURE II-6

SOLVENT EXTRACTION CYCLE

additions for valence adjustment and the use of ion-exchange resin columns. The quantities of waste to be subsequently handled and disposed of, are also reduced. In this analysis, the former unit operation is reviewed for its accident potential.

- Uranium Purification and Recovery. The aqueous uranium strip solution is concentrated and its acidity is adjusted prior to resubjecting it to another partitioning cycle and filtration through a silica gel bed for final concentration loadout as uranyl nitrate solution. The operations remove additional residual fission products and plutonium further ensuring that the uranium solution meets product specifications for reuse in the fuel cycle. The nitrate solution is ultimately shipped as such or converted to UF_6 in an associated facility prior to shipment for reuse in the fuel cycle.

- Plutonium Purification and Recovery. Plutonium in the aqueous stream may be cyclically converted to anionic $Pu(NO_3)_6^{--}$ and sorbed on strong base anion exchange resin while the associated uranium and fission products are washed out in the raffinate. The sorbed plutonium from the last purification cycle is eluted as nitrate solution, concentrated and stored pending conversion to plutonium dioxide for use in plutonium recycle or breeder reaction fuel elements. Alternatively, the aqueous plutonium stream leaving the partitioning column may be reoxidized to the extractable tetravalent state and subjected to additional extraction-scrubbing sequences as described above to further decontaminate the product solution from fission products. The final plutonium nitrate solution, after concentration, is analyzed and stored in geometrically favorable tanks until it is converted in the Plutonium

Product Facility to the solid oxide form for storage and/or off-site shipment.

- Organic Solvent Systems. The organic solvent waste streams from the decontamination and partition cycles are washed successively with dilute aqueous solutions of sodium carbonate, followed by nitric acid and neutralized by sodium carbonate to remove organic degradation products by extraction or precipitation. Precipitated solids are removed by filtration. As required, fresh TBP or dilutent is added to maintain the TBP concentration and the total solvent inventory.

- Liquid Waste Treating and Storage. The aqueous raffinate streams from the plutonium and uranium cycles are reprocessed for residual fissile material content by extraction into TBP organic solution which is recycled back to the decontamination cycle for recovery. The aqueous raffinate, essentially depleted of radioactive materials, is then concentrated in the low level waste concentrator. The radioactive waste streams from all the solvent extraction cycles are concentrated in the high-level or low-level waste concentrators to recover nitric acid and water for reuse in the process while reducing the waste volumes to be stored in appropriate waste facilities; miscellaneous waste streams containing salts, minimal fission products and no appreciable fissile material are acidified and concentrated. The concentrator bottoms are stored in appropriate waste tanks for ultimate disposal at a federal repository and the condensed overheads are vaporized to the stack.

- Process Off-Gas Treatment. In order to maintain near-atmospheric internal working pressures, with very few exceptions, all of the process equipment pieces - vessels, extractors, condensers, etc. - are vented to one or more process vent systems. The vent gases are treated by scrubbing with circulating mercuric nitrate solution to remove radioactive iodine, then treated in an absorber to convert nitrogen oxides to nitric acid suitable for recycling. The dissolver off-gas and vessel off-gas streams are combined and passed through a second decontamination train which includes an iodine scrubber unit, iodine adsorber beds and a series of high efficiency filter banks before being released to the stack. Thus, the relatively small amounts of radioactivity including most of the remaining radioiodine are removed from the vent gases prior to discharge, via a stack, to the atmosphere. The vent gases will probably contain most of the tritium as a result of operations in the acid recovery system. Virtually all of the tritium from the original fuel follows the water phase, as tritiated water, to the waste system and thence to the process condensates derived during evaporation-acid recovery operations. In early process designs, this material was discharged to the environment following evaporative and chemical treatments, for removal of entrained non-volatile activity. Current design approaches recycle process condensates in the process, as process water, with evaporation of a small-volume remainder of the condensate to discharge tritium to the atmosphere via the off-gas system and the stack.

The gaseous wastes from a reprocessing plant include a ventilation air, which is indirectly exposed to the process, besides the process off-gases which are directly exposed to the process and which contain some volatile radionuclides. Both systems are shown by Figure II-7, page 27. In this block diagram, the process building is shown schematically as the large block on the left. The building operating spaces are divided into four zones of ventilation control, with corresponding increasing levels of potential contamination by radioactivity, and increasing degrees of access control requirements. As plotted on the diagram, the building ventilation system is engineered to maintain pressure differentials between zones such that the air flow is always toward the zone of greater potential contamination. The Zone 1 spaces, shown at a slightly positive air pressure, include offices, lunchrooms, etc., and no special radiological control measures are required. Examples of Zone 2 spaces include parts of the analytical laboratories. The potential for radioactive contamination is low, but controlled access is required. Zone 3 spaces include plutonium product loadout spaces, other spaces of the analytical laboratories, etc. Access is supervised. Zone 4 spaces are those which are expected to be routinely contaminated, such as process cells. The negative pressure in Zone 4 is typically an inch of water.

A relatively large volume of air continuously flows through the process building zones, through an efficient filtration system, and to the atmosphere via a stack.

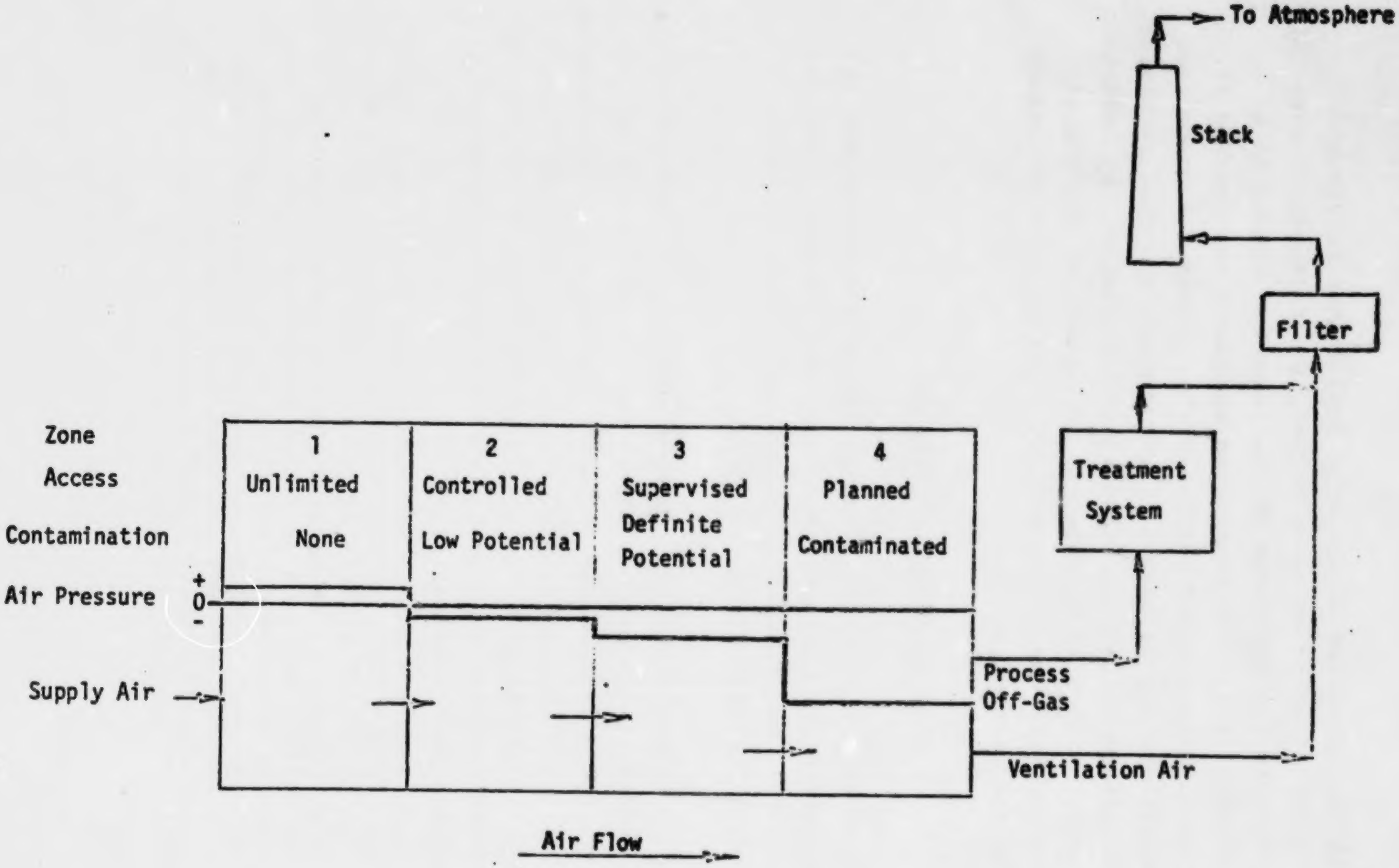

Zone
Access
Contamination
Air Pressure
+
0
-
Supply Air
1
Unlimited
None
2
Controlled
Low Potential
3
Supervised
Definite
Potential
4
Planned
Contaminated
Process
Off-Gas
Ventilation Air
Treatment
System
Filter
Stack
To Atmosphere
Air Flow

FIGURE II-7
OFF-GAS AND VENTILATION WASTES

With the exception of incoming fuel shipments and the transfer of solid scrap to burial, all radioactive material handled in the operation of the plant is located in an exclusion area. All processing equipment and systems for processing irradiated fuel elements, except for waste tank facilities, may be housed in the process and fuel receiving and storage station building as exemplified in the isometric drawing, Figure II-2, page 10. A summary of the process functions and chemical reactions involved in reprocessing is given in Appendix A.

4. Radioactivity Confinement

The general concept of radioactivity confinement used in the generic plant analysis has been used for many years and has performed reliably in plants processing LWR fuels decayed for half a year or more.

The reprocessing facility design and process flow operations include multiple methods and systems to insure control of radioactive material so that in-plant and off-site exposures exceeding allowable limits, are not experienced throughout the life of the plant under normal operating conditions and routine maintenance. Confinement is based upon the use of one or more of the following:

- Barriers to confine radioactive materials in the plant which include process piping and equipment, building structures, casks and storage tanks.
- Channeling to confine the flow of fluid within fixed barriers or direct it through fixed barriers by pressure differentials or by gravity. Channeling is used to direct potentially contaminated fluids away from plant personnel and/or points of uncontrolled release toward a treatment system or point of controlled release.

- Treatment to remove radioactive material from fluids or gases discharged to the environment so that established limits are not exceeded.
- Burial to confine certain non-mobile radioactive solid wastes within the site boundaries.

The methods used, depending upon the mobility, quantity, type and intensity of the associated radioactivity for the unit operation involved, are shown in Table II-3, page 30.

Accidental radioactive releases, as measured by an increase in off-site radiation level, could follow three pathways: release to liquid effluents, release to ground water and release to the atmosphere. The confinement and ventilation systems in fuel reprocessing plants remove particulates of non-volatiles dispersed under accidental conditions and liquid releases to such an extent that off-site environmental risks are dominated by airborne releases of volatile and semi-volatile materials only. Therefore, the airborne release pathway was the major one considered for evaluations of the off-site environmental risks from potential accidents in the plant operations. Secondary emphasis was placed on liquid releases to the ground. A radioactive material flow diagram for a reprocessing plant, as exemplified by BNFP, is shown schematically in Figure II-8, page 31.

5. Administrative Controls

The operation of a radiochemical separations plant is in most respects like the operation of any large chemical plant except for the complications introduced by the radioactive nature of the process materials. Working with radioactive materials necessitates adherence to extensive government regulations for their control. These are found

TABLE II-3

CONFINEMENT

	Confinement Method Used			
	Barriers		Channeling	Treatment
	Single	Multiple		
a. Pool Water	x		x	x
b. Fuel Elements-Undamaged		x		
c. Fuel Elements-Damaged	x		x	x
Process				
d. Fluids		x	x	x
e. Solids	x		x	x
Product				
f. Uranium (Storage & Shipping)	x			
g. Plutonium & Neptunium		x	x	x
* h. Plutonium & Neptunium (Storage & Shipping)		x		
Waste				
i. High Level Liquid		x	x	x
* j. Intermediate Liquid		x	x	x
* k. Low Level Liquid with Tritium	x		x	x
l. Other Low Level Liquid	x		x	x
m. Solids (Hulls & Equipment) in cell	x		x	x
* n. Hulls during Transport		x		
* o. Contaminated Equipment & Solid * Radioactive Wastes during Transport	x(1)	x(1)		

* (1) Number of barriers will be dependent upon the radiation level of the material. If the
* radiation level is high, shielding will be provided during transport.

BNFP RADIOACTIVE MATERIAL FLOW

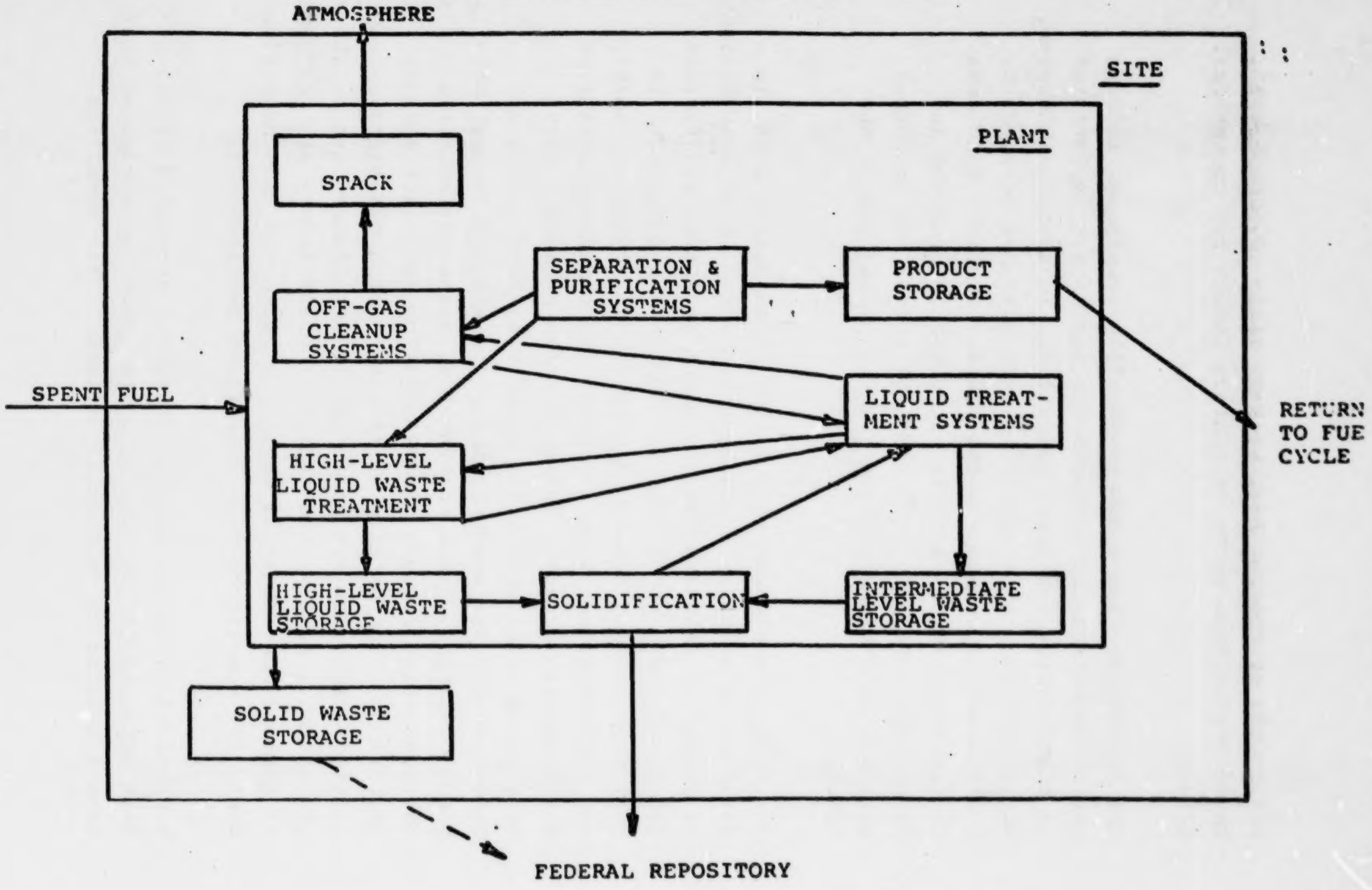

ATMOSPHERE
SITE
PLANT
STACK
OFF-GAS CLEANUP SYSTEMS
SEPARATION & PURIFICATION SYSTEMS
PRODUCT STORAGE
SPENT FUEL
LIQUID TREAT-MENT SYSTEMS
RETURN TO FUE CYCLE
HIGH-LEVEL LIQUID WASTE TREATMENT
HIGH-LEVEL LIQUID WASTE STORAGE
SOLIDIFICATION
INTERMEDIATE LEVEL WASTE STORAGE
SOLID WASTE STORAGE
FEDERAL REPOSITORY

FIGURE II-8

under Code of Federal Regulations, Title 10, Chapter I[9]. These regulations serve as minimum limits for operational safety.

The principles upon which criticality safety and radioactive contamination controls are based, are implemented in plant design and plant operation. In addition, however, administrative responsibilities are assigned to specific individuals or groups, at the plant, for plant functions to assure that the reprocessing plant is operated and maintained under the full range of normal and potential accident conditions without risk to public health and safety.

The daily functioning of the fuel reprocessing plant is governed by an on-site organization which is self-sufficient in regard to assuring public, plant personnel and facility safety on a day-to-day basis. Functional components in this organization include engineering, production, safety and analytical services, employee and community relations and financial groups. These licensed personnel have specific qualifications for overseeing criticality and radiation safety, accountability for special nuclear materials, plant operation, plant maintenance, plant assistance and plant services which include decontamination and waste disposal operations. The on-site staff assure that all safety-related activities are performed in accordance with established procedures. Reviews and audits of plant safety are performed at appropriate intervals both on an internal and independent basis for compliance with prescribed requirements.

A criticality-safety control committee appointed by management establishes the limits on the operating variables that have a bearing on criticality safety. This committee, composed of representatives from all the functioning groups, reviews proposed changes in equipment or in operating procedure. The committee's approval is required before any change is implemented.

Radiation survey inside the plant as well as in the site environs within a radius of up to 50 miles verifies the effectiveness of contamination control.

Through a system of checks and balances among the functional components in carrying out their daily operations of the plant, the maintenance of performance records, the institution of training programs and testing of personnel to assure their ability to discharge safety-related responsibilities and the performance of audits at regular intervals, plant safety under all normal and abnormal operation conditions as well as full compliance with license and regulatory requirements are further assured.

6. Auxiliary Plant Systems and Controls

a. Ventilation and Off-Gas System

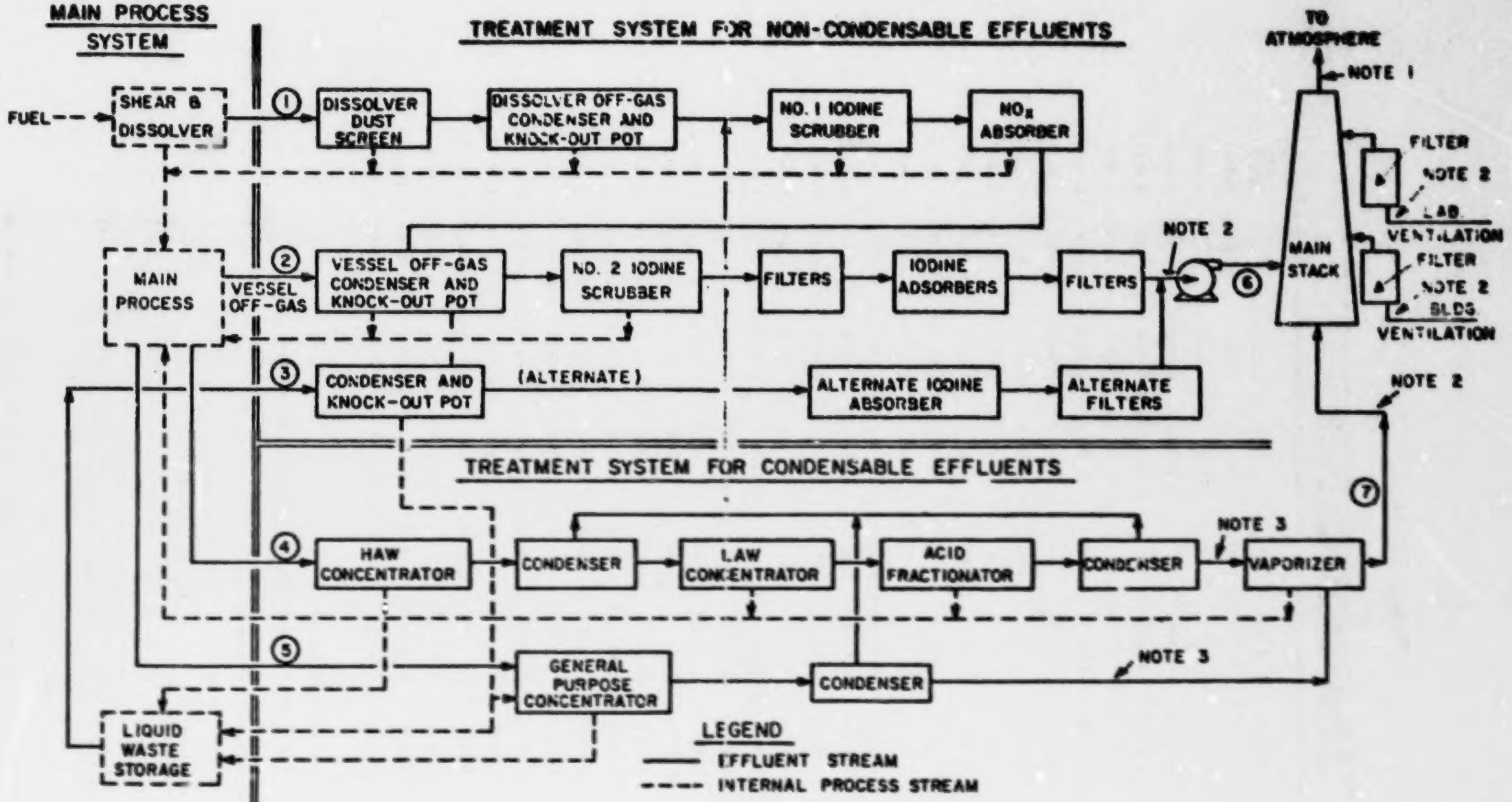
In the generic plant, only gaseous and solid radioactive wastes will eventually leave the site. No radioactive liquid effluents will be exposed to the environment.

The primary effluent that could have an impact on the environment during operation of the facility is the gaseous effluent from the stack. This was discussed briefly in an earlier section and in more detail in the following.

The gaseous effluent from the main stack consists of building ventilation air, vaporized process condensate and the off-gases from the dissolver and vessel off-gas treatment system. The building ventilation exhaust air is routed through at least two stages of high efficiency (HEPA) filters prior to discharge to the atmosphere. Excess process condensate is decontaminated by evaporation and condensation. The decontaminated water is recycled to the process or alternatively may be revaporized and discharged to the atmosphere via the 100-meter tall stack. The dissolver off-gases (DOG) are routed through a treatment train consisting of a condenser, knock-out pot, iodine scrubber and an NO_2 absorber. The NO_2 absorber is designed to recover 70 percent of the NO_x as 45 percent nitric acid. The treated dissolver off-gas, which still contains residual amounts of NO_x and iodine, is further treated through the vessel off-gas treatment system (VOG) which also collects gases vented from various other process or storage vessels. The VOG treatment system consists of a condenser, knock-out pot, iodine scrubber, pre-filter, iodine absorber and a two-stage high efficiency (HEPA) filter. The stack is equipped with samplers, monitors and alarms which identify the amounts of radioactivity in the effluent. The system is illustrated in Figure II-9. The expected radioactive release from use of the treatment systems shown in Figure II-9 in processing high-exposure spent fuel (average annual fuel exposure of 32,000 MWD/MTU at 40 MW/MTU) cooled 180 days from reactor discharge, and processed at the rate of 1500 MTU/year is listed in Table II-4, page 36.

b. Water Supply

Water must be provided for process streams, for makeup water for closed loop cooling systems, for cooling towers


MAIN PROCESS SYSTEM
TREATMENT SYSTEM FOR NON-CONDENSABLE EFFLUENTS
TO ATMOSPHERE
NOTE 1
FUEL
SHEAR & DISSOLVER
①
DISSOLVER DUST SCREEN
DISSOLVER OFF-GAS CONDENSER AND KNOCK-OUT POT
NO. 1 IODINE SCRUBBER
NO$_x$ ABSORBER
FILTER
NOTE 2
LAB. VENTILATION
MAIN PROCESS
VESSEL OFF-GAS
②
VESSEL OFF-GAS CONDENSER AND KNOCK-OUT POT
NO. 2 IODINE SCRUBBER
FILTERS
IODINE ADSORBERS
FILTERS
NOTE 2
⑥
MAIN STACK
FILTER
NOTE 2
BLDG. VENTILATION
③
CONDENSER AND KNOCK-OUT POT
(ALTERNATE)
ALTERNATE IODINE ABSORBER
ALTERNATE FILTERS
NOTE 2
TREATMENT SYSTEM FOR CONDENSABLE EFFLUENTS
⑦
NOTE 3
④
HAW CONCENTRATOR
CONDENSER
LAW CONCENTRATOR
ACID FRACTIONATOR
CONDENSER
VAPORIZER
⑤
GENERAL PURPOSE CONCENTRATOR
CONDENSER
NOTE 3
LIQUID WASTE STORAGE
LEGEND
EFFLUENT STREAM
INTERNAL PROCESS STREAM

NOTES:

1. THIS STREAM IS CONTINUOUSLY SAMPLED FOR RADIOACTIVE CONTENT.
2. THESE STREAMS ARE SAMPLED FOR RADIOACTIVE CONTENT AS REQUIRED.
3. THESE STREAMS ARE CONTINUOUSLY SAMPLED & MONITORED WITH PROVISION FOR DIVERSION & RECYCLE IF REQ'D.

FIGURE II-9

SEPARATIONS FACILITY EFFLUENT TREATMENT SYSTEM[7]

TABLE II -4

ESTIMATED AVERAGE RADIOACTIVE EFFLUENTS
(curies/sec) - Separations Plant Main Stack

Stream No. →	1	2	3	4	5	6	7	Total To Stack
*H-3	3.6E-3	--	--	1.4E-2	--	3.6E-3	1.4E-2	1.8E-2
*Kr-85	4.3E-1	--	--	--	--	4.3E-1	--	4.3E-1
Sr-89	2.0E-6	1.0E-5	2.0E-5	4.1	1.1E-3	6.5E-9	1.1E-9	7.6E-9
*Sr-90	2.3E-6	1.2E-5	2.3E-5	4.8	1.3E-3	7.6E-9	1.3E-9	8.8E-9
Y-90	2.3E-6	1.2E-5	2.3E-5	4.8	1.3E-3	7.6E-9	1.3E-9	8.8E-9
Y-91	4.2E-6	2.1E-5	4.2E-5	8.7	2.4E-3	1.4E-8	2.3E-9	1.6E-8
Zr-95	7.4E-6	3.8E-5	7.4E-5	15.4	4.2E-3	2.4E-8	4.0E-9	2.8E-8
Nb-95	1.4E-5	7.1E-5	1.4E-4	29	7.9E-3	4.6E-8	7.6E-9	5.3E-8
Ru-103	2.2E-6	1.1E-5	2.2E-5	4.6	1.2E-3	7.2E-9	1.2E-9	8.4E-9
Ru-106	1.3E-5	6.7E-5	1.3E-4	28	7.4E-3	4.3E-8	7.2E-9	5.0E-8
*I-129	1.4E-6	--	--	8.3E-8	6.1E-8	1.4E-10	1.1E-9	1.5E-9
*I-131	1.1E-5	--	--	5.3E-7	5.3E-7	1.1E-9	1.1E-8	1.2E-8
*Cs-134	5.0E-6	2.6E-5	5.0E-5	11	2.9E-3	1.7E-8	2.8E-9	1.9E-8
*Cs-137	3.2E-6	1.6E-5	3.2E-5	6.7	1.8E-3	1.0E-8	1.7E-9	1.2E-8
Ce-141	1.4E-6	7.3E-5	1.4E-5	3.0	8.1E-4	4.7E-9	7.8E-10	5.5E-9
Ce-144	2.2E-5	1.1E-4	2.2E-4	46	1.2E-2	7.2E-8	1.2E-8	8.4E-8
Pm-147	4.0E-6	2.0E-5	4.0E-5	8.3	2.2E-3	1.3E-8	2.2E-9	1.5E-8
U-234	4.2E-11	2.4E-10	2.2E-12	4.4E-11	2.2E-12	4.4E-14	2.2E-14	7.0E-14
U-235	4.2E-13	2.4E-12	2.2E-14	4.4E-9	2.2E-10	4.4E-16	2.2E-16	7.0E-16
U-238	9.9E-12	5.8E-11	5.4E-13	1.1E-7	5.4E-9	1.4E-14	5.4E-15	2.0E-14
*Pu-238	1.0E-7	2.8E-7	1.1E-8	2.3E-3	5.6E-5	7.9E-11	5.3E-11	1.3E-10
*Pu-239	9.0E-9	2.5E-8	9.9E-10	2.0E-4	5.0E-6	7.3E-12	4.8E-12	1.2E-11
*Pu-240	1.6E-8	4.6E-8	1.8E-9	3.6E-4	9.1E-6	1.2E-11	8.7E-12	2.1E-11
*Pu-241	4.2E-6	1.2E-5	4.5E-7	9.1E-2	2.3E-3	3.3E-9	2.2E-9	5.5E-9
Pu-242	8.6E-11	2.4E-10	9.5E-12	1.9E-6	4.8E-8	6.8E-14	4.6E-14	1.1E-13
*Am-241	8.7E-9	4.3E-8	8.7E-8	1.4E-2	4.3E-6	2.8E-11	4.3E-12	3.2E-11
Am-242	1.7E-10	8.4E-10	1.7E-9	2.7E-4	8.4E-8	5.5E-13	8.4E-14	6.3E-13
*Cm-242	1.0E-6	5.1E-6	1.0E-5	1.7	5.1E-4	3.2E-9	5.1E-10	3.7E-9
Cm-243	6.7E-10	3.4E-9	6.7E-9	1.1E-3	3.4E-7	2.1E-12	3.4E-13	2.5E-12
*Cm-244	1.1E-7	5.4E-7	1.1E-6	1.8E-1	5.4E-5	3.4E-10	5.4E-11	3.9E-10

*Quantitatively and Biologically Significant.

Short lived decay products are also present in equilibrium quantities.

For Peak Releases, multiply Krypton and Iodine values in Stream #7 by 2.2, and remaining values in Streams #7 and #8 by 1.2.

[1]ACTIVITY BASIS:
SPECIFIC POWER - 40 kw/kg
COOLING TIME - 180 days
FUEL ACTIVITY - 32,000 MWD/MTU
PRODUCTION RATE - 1,500 MTU/Yr.

RELEASE BASIS:
100% Kr-85 & H-3 Released
Iodine Decontamination Factor = 1,000
DOG/VOG Decontamination Factor = 10 and Filter Efficiency = 99.8%
HAW/Vaporizer Decontamination Factor = 10^{12}
GPF/Vaporizer Decontamination Factor - 10^6
Pu Product Recovery = 99%
Pu Loss to Waste = 1%

---

[1] Average source during reprocessing is assumed to be less than peak. Values are based on instantaneous rate averaged over a calendar year.

NOTE: 1E-10 = 1 x 10^{-10}

and for the fuel storage pools. In addition, water must be provided as backup for the plant fire water system, emergency cooling for the HLLW storage facilities, for the emergency utility area, and for the 76°F cooling loop.

In the generic plant, it is assumed that water will be available from deep wells on the site. (At BNFP, water is obtained from three deep wells. Normal cooling is by circulation through the cooling tower but the wells can supply straight-through cooling emptying into a pond which serves as an emergency reservoir. The BNFP pond is a 15-acre pond having a capacity of 60 million gallons of water.)[7] Recirculation, for several months, with such a pond is possible by means of an emergency diesel-powered pump. Redundant pumps are connected to separate emergency busses with automatic start features as a back-up safety measure.

c. On-Site Electrical Power

The loss of on-site electrical power, if sustained, could lead to an unsafe plant condition. To minimize this possibility, consideration is given to a high reliability source of off-site power. Should off-site power fail, standby diesel powered generators will come on and assume priority loads. These and day-tank fuel supplies are located in structures designed to meet earthquake and tornado criteria[5,6,7]. An uninterruptable source of ac power is used to supply power to the process control equipment and to provide control for starting the diesel generators. A 24V dc battery supply provides a highly reliable supply of power for monitors and some control functions.

d. Compressed Air Supply

Oil free compressed air is required for process control, instrumentation and starting the emergency diesel powered generators. It is used for air lifts, air pulsers for the extraction column operations, for air circulation as in the high-level liquid waste storage tanks, and for air purging of radiolytic hydrogen concentrations, generated during plant operation, to prevent formation of potentially explosive concentrations.

The generic plant is provided with two air compressors to provide a doubly redundant supply of air. These compressors are provided with pressure relief valves as a precaution in the event of failure of the pressure cut-off switch. Accumulator tanks located at critical locations about the plant are equipped with reverse flow check valves in their feed lines so compressor failure or pipe failure will not necessarily result in immediate pressure failure. As an example, the emergency diesels have accumulator tanks capable of 5 starts without resupply.

e. Steam Supply

Steam is used for process transfer jets, process heating, space heating, steam turbines, decontamination, deaerator heating and stripping, and yard steam tracing.

In general, it is not a critical service except for two aspects: the high-level liquid waste transfer and the control of steam pressure (100 psig) and hence the temperature (270°F). To inhibit the possibility of a "red-oil" explosion which requires approaching a threshold temperature as one of the conditions, the steam pressure is limited to 40 psig in the solvent extraction operations and subsequent processing of the effluent streams containing organic solvents. Redundant pressure regulators are used to maintain the maximum allowable process temperatures during the various unit operations.

The generic plant, having a large quantity of high-level liquid waste, has an emergency steam generator for emergency liquid waste transfer which is designed to earthquake acceleration criteria and is housed in an enclosure built to tornado criteria.

III. FAULT TREE CONSTRUCTION OF ACCIDENT SEQUENCES

1. Background

Quantitative safety analysis has been developing rapidly due to the requirements of the space program and to the growth in size and numbers of nuclear power reactors. The techniques which have developed in these disciplines are used in this study for the analysis of hypothetical nuclear fuel reprocessing accidents and therefore a brief review of the development of reactor safety analysis will be presented.

The accident potential in nuclear power reactors has been recognized for some time. Safety analysis of nuclear power reactors has been approached by designing them for protection against the maximum credible accident (MCA). Later, in a desire for more realism, the design basis accident(s) (DBA) was defined and mitigated against. Currently there is a desire to present accidents on a numerical risk basis so that comparisons can be made with risks already accepted by society. Early work in probabilistic reactor accident assessment was done by Mulvihill[12], Garrick et al[13], Farmer[14] and Otway and Erdmann[15]. All of these authors except Farmer used fault tree analysis for the calculation of accident probabilities. This work is actively continuing under AEC sponsorship. To date, the most complete descriptions of reactor risk are provided by Otway and Erdmann[15] and Starr[16] who place these risks in perspective by comparisons with other social risks. The present work provides a similar safety analysis for a generic nuclear fuel reprocessing plant.

The central problem of probabilistic safety analysis is the representation of a complex system such as a fuel reprocessing plant in a form suitable for safety analysis. A plant is represented by construction, plumbing, electrical drawings, process flow charts, etc.; these must be synthesized into a unified description of the plant accident spectrum which is suitable for probabilistic analysis. This synthesis can be done through the use of fault trees.

Engineering for safety is not a new concept. Common examples are pressure relief valves on a boiler, or the safety on a loaded firearm. In both of these examples two simultaneous events must occur for an accident to result. Early safety engineering was usually done after the fact of an accident to prevent its recurrence. And in fact, the beginning of probabilistic safety was concerned with analyzing after-the-fact missile misfirings. Nuclear safety engineering has introduced a new aspect; namely, the calculation of the occurrence probability of accidents that in many cases have never occurred, or have never occurred with the safety system being analyzed.

Normally, reasoning proceeds from cause to effect; this process, when applied to safety, is called failure mode and effects analysis. This method proceeds by a series of "what if" statements to the final undesired event but generates many final results that are not of particular interest.

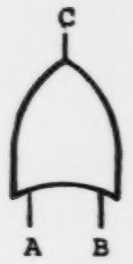
Fault tree analysis begins with a final result that is significant and proceeds back through the system identifying causes. In a certain sense, time is going backwards in the logic of developing the tree. In this manner, many extraneous paths that would be generated by a failure-mode and effects analysis are eliminated and only those paths that lead to the top event are generated.

The logical structure dictated by the form of the fault tree prescribes the manner in which the probabilities must be combined to yield both the probability of the top node (final hazard event), and the probability of individual event sequences.

The overall accuracy of event sequence prediction, therefore, depends directly upon both the availability and quality of the basic probability data .

2. Fault Tree Symbolic Language

Fault tree analysis is basically a two-state Boolean logic and as such uses the operation of logical addition often referred to as logical "or" ("union").



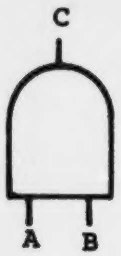
A + B = C

Truth Table

A	B	C
1	0	1
0	1	1
0	0	0
1	1	1

Thus if A or B have a true input, the output is true. In the case of fault trees, true corresponds to failure. If either of the inputs A or B are in the failed state the gate output is in the failed state.

Another logical operation is logical multiplication and is referred to as logical "and" ("intersection").



$$A \cdot B = C$$

Truth Table

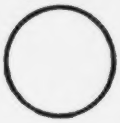
A	B	C
1	0	0
0	1	0
0	0	0
1	1	1

Thus if A and B are true, the output is true and two safety systems must simultaneously fail to get system failure. In the earlier example of a loaded firearm, the pulling of the trigger, and the safety being off must both occur for the weapon to fire.

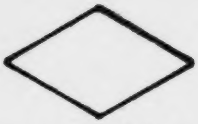
Other logical operations have been defined and are used by safety analysts. In this report, care has been used to avoid other operations because: they usually can be represented by combinations of "and" and "or" gates, for reasons of clarity and because these operations are the only ones that can be directly treated in existing fault tree computer programs.

The accident events are represented by a square which describes the event logically represented by a gate.

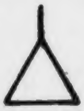
A circle is used to represent an event for which probability-of-occurrence data is available. Hence it represents a terminal event requiring no further development.



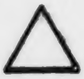
A diamond is used to represent an event which is not further developed because it is not believed to be of significance or adequate information is unavailable.



The triangle is used as a linking symbol.



LINK IN



LINK OUT

A triangle, always to the right of an event rectangle, represents a linking out and indicates that the whole tree should be transferred over to the linking-in connection which is always at bottom center of the link-in point. The linking operation is used to avoid repetitious drawing and space limitations. In this report, the link-in rectangle may give information by further specifying the tree being linked-into. In this way a general repeated tree is particularized for a specific application. This procedure is similar to a subroutine calling sequence which transfers parameters to the subroutine calculation.

3. Fault Tree Construction for a Nuclear Fuel Reprocessing Plant

A chemical plant is a very complex entity in both its architecture and processing operations. Correspondingly a fault tree modeling the safety analysis of the plant is also highly complex. To systematize and maintain control over the completion of the safety modeling a procedure for fault tree construction known as Leak Path Analysis was used.[17] Briefly, the method enumerates the intersection of barrier failures encountered in tracing all paths from radioactive sources to the environment. The union of these Leak Paths forms the top event - the uncontrolled release of radiation (URR) to the environment. This very large equation is both unwieldy and contains redundancies but when factored into most compact form (terms appearing only once) or as close thereto as can be achieved, it is suitable for conversion to a fault tree of the plant. This is performed by the replacement of intersection operations by the fault tree symbol AND and similarly union by OR.

The top of a fault tree constructed in this manner is shown in figure III-1. The modes of release are air and liquid pathways. Solid pathways were not considered.

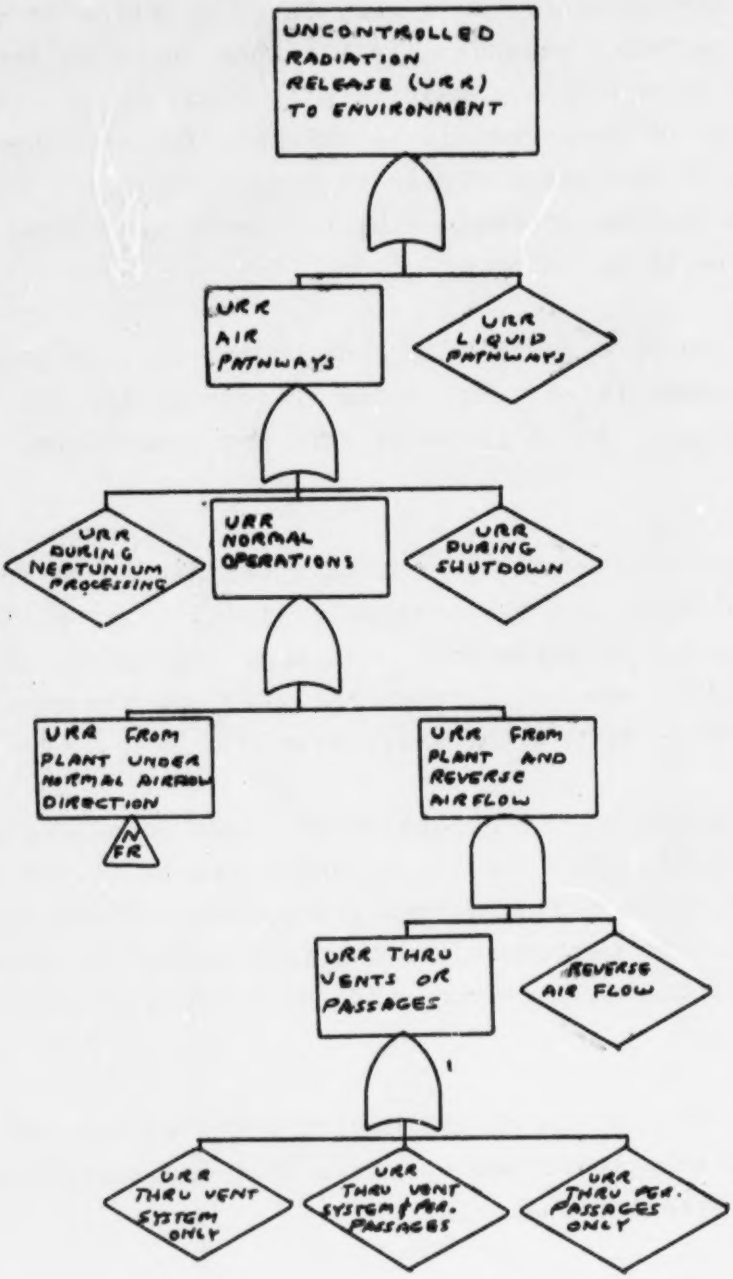
The release by liquid pathways was developed in a series of fault trees for qualitative evaluation of critical paths but based on historical data from high level liquid waste storage[18] was not further analysed and the event is now "diamond-off" on the fault tree.

Similarly, an investigation of risks under neptunium processing failed to identify consequences as severe as those which could occur during normal processing. Since neptunium processing is performed about once a year, the probability is correspondingly reduced and the risks are not further considered.

An investigation of URR during shutdown did not reveal risks comparable to normal operation and was therefore excluded from detailed analysis.

Figure III-1

Generic Reprocessing Plant: Top of Tree


UNCONTROLLED RADIATION RELEASE (URR) TO ENVIRONMENT
URR AIR PATHWAYS
URR LIQUID PATHWAYS
URR DURING NEPTUNIUM PROCESSING
URR NORMAL OPERATIONS
URR DURING SHUTDOWN
URR FROM PLANT UNDER NORMAL AIRFLOW DIRECTION
NFR
URR FROM PLANT AND REVERSE AIRFLOW
URR THRU VENTS OR PASSAGES
REVERSE AIR FLOW
URR THRU VENT SYSTEM ONLY
URR THRU VENT SYSTEM & PER. PASSAGES
URR THRU PER. PASSAGES ONLY

Release by air pathways under normal plant operation is divided into normal and reverse air flow because reverse air flow allows particulate from cells such as the PPC to impact the environment without any high efficiency filtering. In the investigation it was found that reverse air flow cannot occur by equipment failure alone. There is one exception, however, the air pulser can in principle result in a higher pressure in the process cells than outside the building. Considering the reverse flow ventilation damper, the redundant protection for maintaining normal air flow, the interlock on the pulser and the volume of building and site of air flow, this mechanism was determined to be ineffective and was not pursued. Fault trees were constructed for the reverse air flow diamonded events shown in Figure III-1 but are not included because they were not numerically evaluated and void in the risk assessment.

A tornado is a natural event that can result in reverse air flow from its depressurization and this is included in the tornado analysis. Because of the uncertainty of failure modes under tornado conditions, little value is derived from the reverse flow fault trees.

The fault tree development continues with the linking tree NFR (figure III-2). The failure probability of 10^{-2}/yr or 10^{-3}/yr for single and double HEPA filters respectively are from references 19 and 20. This is the probability of the filter having a particulate transmission greater than design specification. Clogging of the filters was not considered to be a failure mode for present purposes.

The event URR in RPC is developed in figure III-3. The releases from this are further divided into those due to

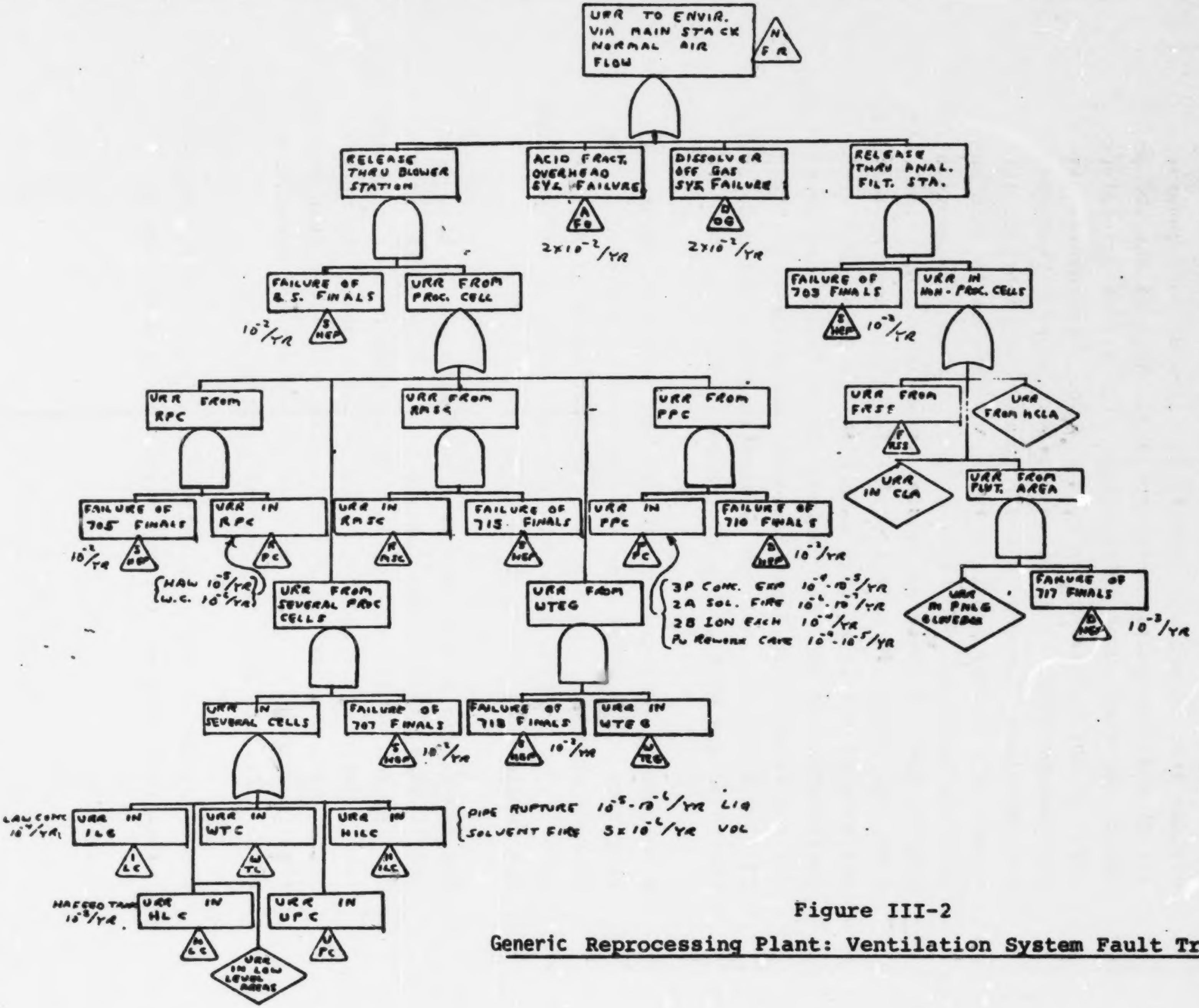

URR TO ENVIR. VIA MAIN STACK NORMAL AIR FLOW
N ER
RELEASE THRU BLOWER STATION
ACID FRACT. OVERHEAD SYS. FAILURE
A FO
2×10^{-2}/YR
DISSOLVER OFF GAS SYS. FAILURE
D OG
2×10^{-2}/YR
RELEASE THRU ANAL. FILT. STA.
FAILURE OF B.S. FINALS
S HEP
10^{-2}/YR
URR FROM PROC. CELL
FAILURE OF 703 FINALS
S HEP
10^{-3}/YR
URR IN NON-PROC. CELLS
URR FROM RPC
URR FROM RMSC
URR FROM PPC
URR FROM FRSF
F RSS
URR FROM HCLA
URR IN CLA
URR FROM PLUT. AREA
FAILURE OF 705 FINALS
S HEP
10^{-2}/YR
URR IN RPC
R PC
HAW 10^{-5}/YR
W.C. 10^{-6}/YR
URR IN RMSC
R MSC
FAILURE OF 715 FINALS
S HEP
URR IN PPC
P PC
3P CONC. EXP 10^{-4}-10^{-5}/YR
2A SOL. FIRE 10^{-6}-10^{-7}/YR
2B ION EXCH 10^{-4}/YR
Pu REWORK CAVE 10^{-4}-10^{-5}/YR
FAILURE OF 710 FINALS
D HEP
10^{-3}/YR
URR IN PNLG GLOVEBOX
FAILURE OF 717 FINALS
D HEP
10^{-3}/YR
URR FROM SEVERAL PROC CELLS
URR FROM UTEG
URR IN SEVERAL CELLS
FAILURE OF 707 FINALS
S HEP
10^{-2}/YR
FAILURE OF 718 FINALS
S HEP
10^{-2}/YR
URR IN UTEG
U TEG
LAW CONC 10^{-4}/YR
URR IN ILC
I LC
URR IN WTC
W TL
URR IN HILC
H ILC
PIPE RUPTURE 10^{-5}-10^{-6}/YR LIQ
SOLVENT FIRE 5×10^{-6}/YR VOL
HAFEED TANK 10^{-3}/YR
URR IN HLC
H LC
URR IN UPC
U PC
URR IN LOW LEVEL AREAS

Figure III-2

Generic Reprocessing Plant: Ventilation System Fault Tree

LIST OF EQUIPMENT

10-C-101	#1 DISSOLVER
10-C-102	#2 DISSOLVER
10-C-103	#3 DISSOLVER
41-D-444	HAW HEAD POT
10-E-132	#1 DISSOLVER HEATER/COOLER
10-E-133	#2 DISSOLVER HEATER/COOLER
10-E-134	#3 DISSOLVER HEATER/COOLER
41-E-401	HAW CONCENTRATOR REBOILER
41-E-460	HAW CONCENTRATOR
10-E-154	#1 DISSOLVER SCREEN POT
10-E-155	#2 DISSOLVER SCREEN POT
10-E-156	#3 DISSOLVER SCREEN POT
09-R-001A/B	MAINTENANCE CEILING PLUG
09-R-001C/D	MAINTENANCE CEILING PLUG
09-D-501	SHIELDING DOOR
06-T-015	FUEL TRANSFER CONVEYOR
06-T-016	FUEL TRANSFER CONVEYOR DRIVE
12-T-101	TRANSVERSE FUEL PUSHER
12-T-102	MAINTENANCE CRANE 35-TON
12-T-103	BASKET HANDLING CRANE
12-T-104	POWER MANIPULATOR
12-T-106A/B	DISSOLVER BASKET STORAGE RACK
13-T-107	FUEL TRANSFER TABLE
46-T-401	DISSOLVER BASKET DUMPER
46-T-403	HULL MONITOR
46-T-411	SUMP JET BASE
46-T-412	SUMP JET BASE
80-T-842	PASS THROUGH PORT
07-V-004	SHIELDING WINDOW
07-V-005	SHIELDING WINDOW
07-V-006	SHIELDING WINDOW
07-V-008	SHIELDING WINDOW
07-V-010A/E	PERISCOPE SLEEVE & PLUG
07-V-010B/F	DITTO
07-V-010C/G	DITTO
07-V-010D/H	DITTO
07-V-021	SHIELDING WINDOW
14-T-101	SHEAR
14-T-104	SHEAR SUPPORT FRAME
14-T-105	SHEAR FEED MAGAZINE
14-T-106	SHEAR MAG. TRANSITION SECT.
14-T-107	SHEAR FEED CHAIN & MAGAZINE
14-T-108	SHEAR FEED RAM DRIVE UNIT
15-T-152	DIVERTER
15-T-153	DIVERTER DRIVE UNIT
19-T-101	FUEL TRANSFER HATCH

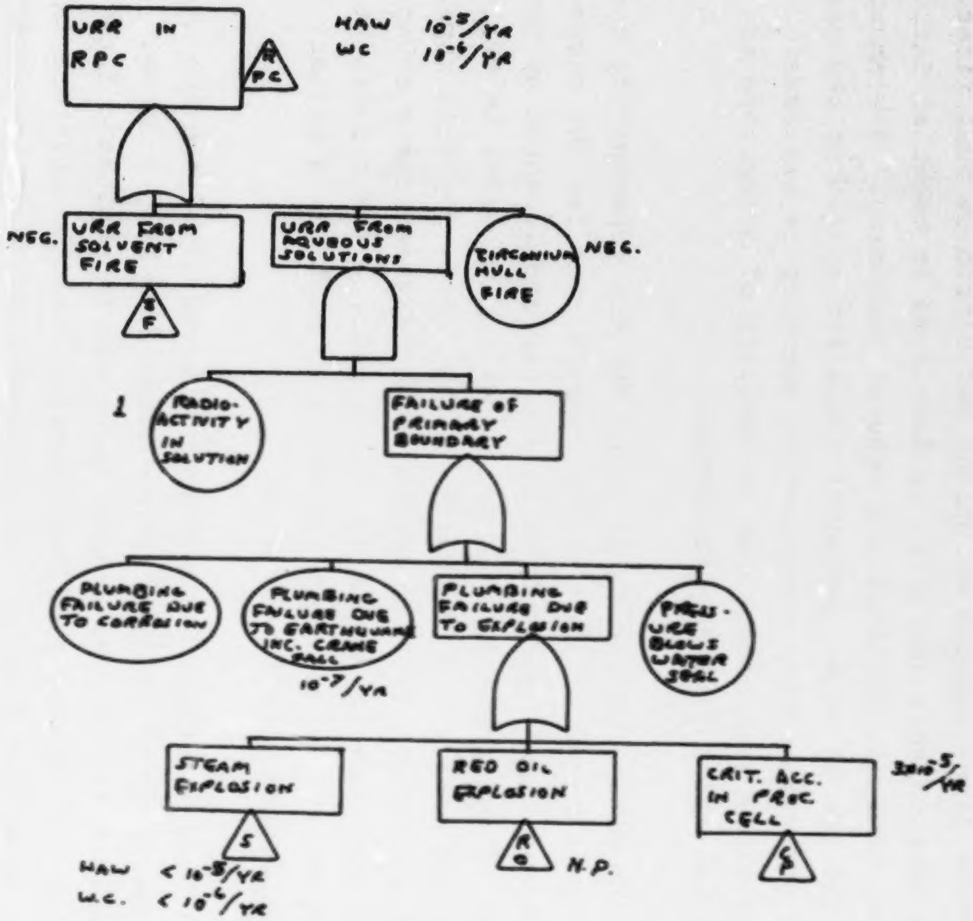

URR IN RPC
R PC
HAW 10^{-5}/YR
WC 10^{-6}/YR
NEG.
URR FROM SOLVENT FIRE
S F
URR FROM AQUEOUS SOLUTIONS
ZIRCONIUM HULL FIRE
NEG.
1
RADIO-ACTIVITY IN SOLUTION
FAILURE OF PRIMARY BOUNDARY
PLUMBING FAILURE DUE TO CORROSION
PLUMBING FAILURE DUE TO EARTHQUAKE INC. CRANE FALL
10^{-7}/YR
PLUMBING FAILURE DUE TO EXPLOSION
PRESS-URE BLOWS WATER SEAL
STEAM EXPLOSION
S
HAW $< 10^{-5}$/YR
W.C. $< 10^{-6}$/YR
RED OIL EXPLOSION
R O
N.P.
CRIT. ACC. IN PROC. CELL
C P
3×10^{-5}/YR

Figure III-3

Generic Reprocessing Plant:

Fault Tree For URR in the Remote Process Cell solvent fire, aqueous solutions and zirconium hull fires. The aqueous solution spill is developed to material failure from chemical, physical and natural phenomena. Statement of the use of water (aqueous) seals for confining the gaseous products within the plumbing boundary is expressed explicitly even though the probability of occurrence is one for about 1 psig overpressure.

In the design of the fault trees for the reprocessing plant, it was decided to present the generic tree for the occurrences in that cell. This is further supplemented by the list of equipment present in the cell. In order to evaluate the accident probability in that cell, repeated applications of the tree to encompass all the equipment given in the list must be performed. This reduces the repetitive complexity of the trees and helps to maintain a better perspective over the details.

The data base used in evaluating figure III-3 and the other trees included in this report comes from many sources. It should first be stated that there are no evaluated data bases for nuclear fuel reprocessing plants, therefore failure rate data from similar equipment in similar environments must be used.

Probably the best source of component data would be from chemical plants using solutions of similar acidity. Accidents are reported to the Manufacturing Chemists Association[21] but there is no reporting of component failure and there are no compilations of this data.

Some component failures and all accidents are reported to the USAEC but there is no systematic compilation of this data. The body of this data is for nuclear reactors, an example of which is a recent analysis of pipe rupture occurrences. Using data reported in 1972, it is found that the log mean pipe failure rate for PWR and BWRs is

$$1.6 \times 10^{-5}/\text{yr-ft}.$$

Similar data is available in the General Electric Pipe Rupture Study series.[22] The report GEAP-10207-25 contains pipe break data for both nuclear and fossil power plants.

Anyakora et al[23] have published instrument failure rate data on chemical plants in Great Britain in three environmental categories. Their work is summarized in a compilation by Powers and Tompkins.[24] Additional British data is contained in the useful text by Bourne and Green[25] and the article by Bourne.[26]

Data on U.S. instruments can be found in MIL-HDBK-217A[27] and in the recent IEEE survey of industrial plants.[28] Data on the reliability of fire prevention systems may be found in the paper by Miller.[29]

General collections embracing electrical, instrumentation and plumbing failure rates are found in the publications by Garrick et al[30] and Atomics International memo.[31] A very useful recent nuclear plant reliability evaluation is that due to Erdmann et al.[32]

The probabilities for natural phenomena are quite uncertain and subject to the siting of the generic plant. A probability of 10^{-7}/yr is estimated for exceeding the design basis earthquake by a factor of 2 in acceleration. This is based upon an unpublished study prepared in connection with an environmental impact analysis for a plant in Richland, Washington. The probability of 10^{-5}/yr for a design basis tornado is from an unpublished study of a plant in Oklahoma.

Using these data, the fault tree of III-3 can be evaluated. The linking tree steam explosion needed for the bottom left hand event is presented in figure III-4. This steam explosion is presented as the intersection of pressure buildup, the failure of pressure relief devices and operator failure. In some portions of the plant where aqueous seals are used, a damaging explosion is not possible. The seal will blow and some of the entrapped radioactivity will constitute a small release within that cell.

Figure III-5, a red oil explosion shows the elements that must be present to result in such a reaction: the presence of heavy ions, excess acidity, organic solvent, excessive temperature and the failure of the operator to correct the upset condition. These events are developed into subsidiary trees as required.

Figure III-6 shows the fault tree for a criticality accident. This is developed in a general way and some care must be exercised in applying it to assure that all the failure modes are indeed possible. The valve failures shown were originally intended to be developed into subsidiary trees to include possible common mode problems. Because these were not found, chemical plant failure rate values were used. The failure rate for instrument power used in figure

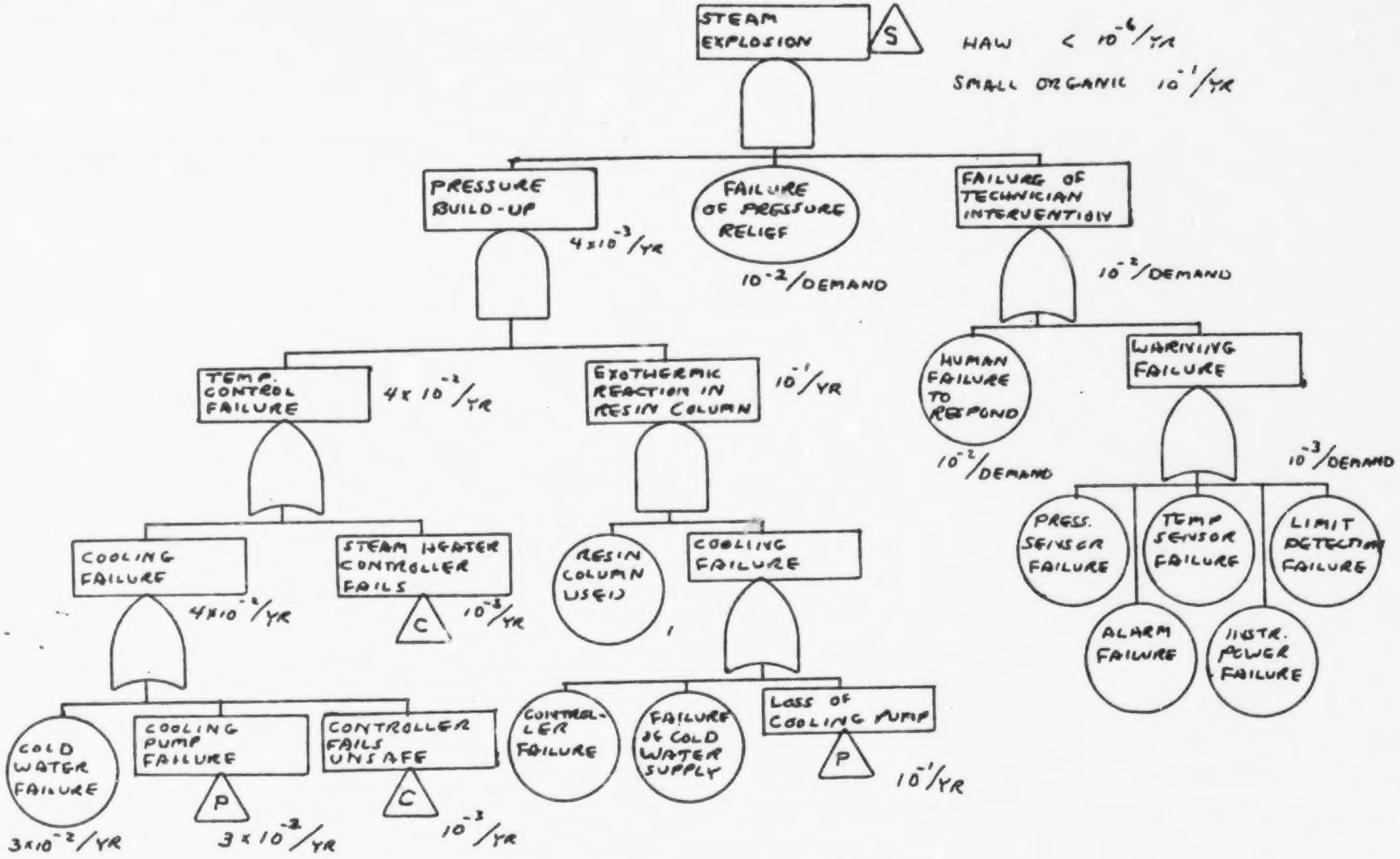

STEAM EXPLOSION
S
HAW < 10^{-6}/YR
SMALL ORGANIC 10^{-1}/YR
PRESSURE BUILD-UP
4×10^{-3}/YR
FAILURE OF PRESSURE RELIEF
10^{-2}/DEMAND
FAILURE OF TECHNICIAN INTERVENTION
10^{-2}/DEMAND
TEMP. CONTROL FAILURE
4×10^{-2}/YR
EXOTHERMIC REACTION IN RESIN COLUMN
10^{-1}/YR
HUMAN FAILURE TO RESPOND
10^{-2}/DEMAND
WARNING FAILURE
10^{-3}/DEMAND
COOLING FAILURE
4×10^{-2}/YR
STEAM HEATER CONTROLLER FAILS
C
10^{-3}/YR
RESIN COLUMN USED
1
COOLING FAILURE
PRESS. SENSOR FAILURE
TEMP SENSOR FAILURE
LIMIT DETECTION FAILURE
ALARM FAILURE
INSTR. POWER FAILURE
COLD WATER FAILURE
3×10^{-2}/YR
COOLING PUMP FAILURE
P
3×10^{-2}/YR
CONTROLLER FAILS UNSAFE
C
10^{-3}/YR
CONTROLLER FAILURE
FAILURE OF COLD WATER SUPPLY
LOSS OF COOLING PUMP
P
10^{-1}/YR

Figure III-4

Generic Reprocessing Plant: Fault Tree for Steam Explosion Accident

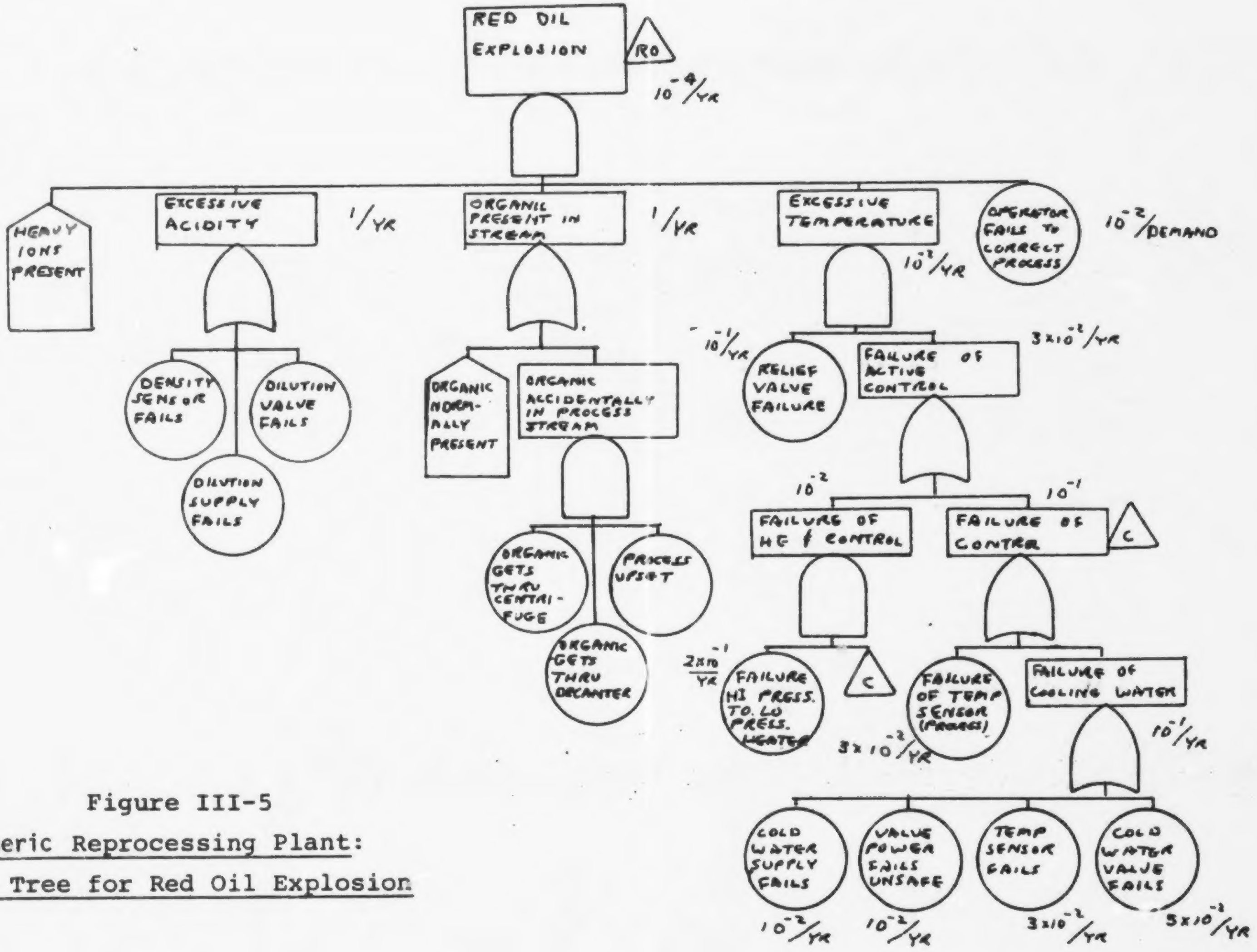

RED OIL EXPLOSION
RO
10^-4/YR
HEAVY IONS PRESENT
EXCESSIVE ACIDITY
1/YR
ORGANIC PRESENT IN STREAM
1/YR
EXCESSIVE TEMPERATURE
10^-2/YR
OPERATOR FAILS TO CORRECT PROCESS
10^-2/DEMAND
DENSITY SENSOR FAILS
DILUTION VALVE FAILS
DILUTION SUPPLY FAILS
ORGANIC NORMALLY PRESENT
ORGANIC ACCIDENTALLY IN PROCESS STREAM
ORGANIC GETS THRU CENTRIFUGE
PROCESS UPSET
ORGANIC GETS THRU DECANTER
10^-1/YR
RELIEF VALVE FAILURE
FAILURE OF ACTIVE CONTROL
3x10^-2/YR
10^-2
FAILURE OF HE ≠ CONTROL
10^-1
FAILURE OF CONTROL
C
2x10^-1/YR
FAILURE HI PRESS. TO LO PRESS. HEATER
C
FAILURE OF TEMP SENSOR (PROCESS)
3x10^-2/YR
FAILURE OF COOLING WATER
10^-1/YR
COLD WATER SUPPLY FAILS
10^-2/YR
VALVE POWER FAILS UNSAFE
10^-2/YR
TEMP SENSOR FAILS
3x10^-2/YR
COLD WATER VALVE FAILS
3x10^-2/YR

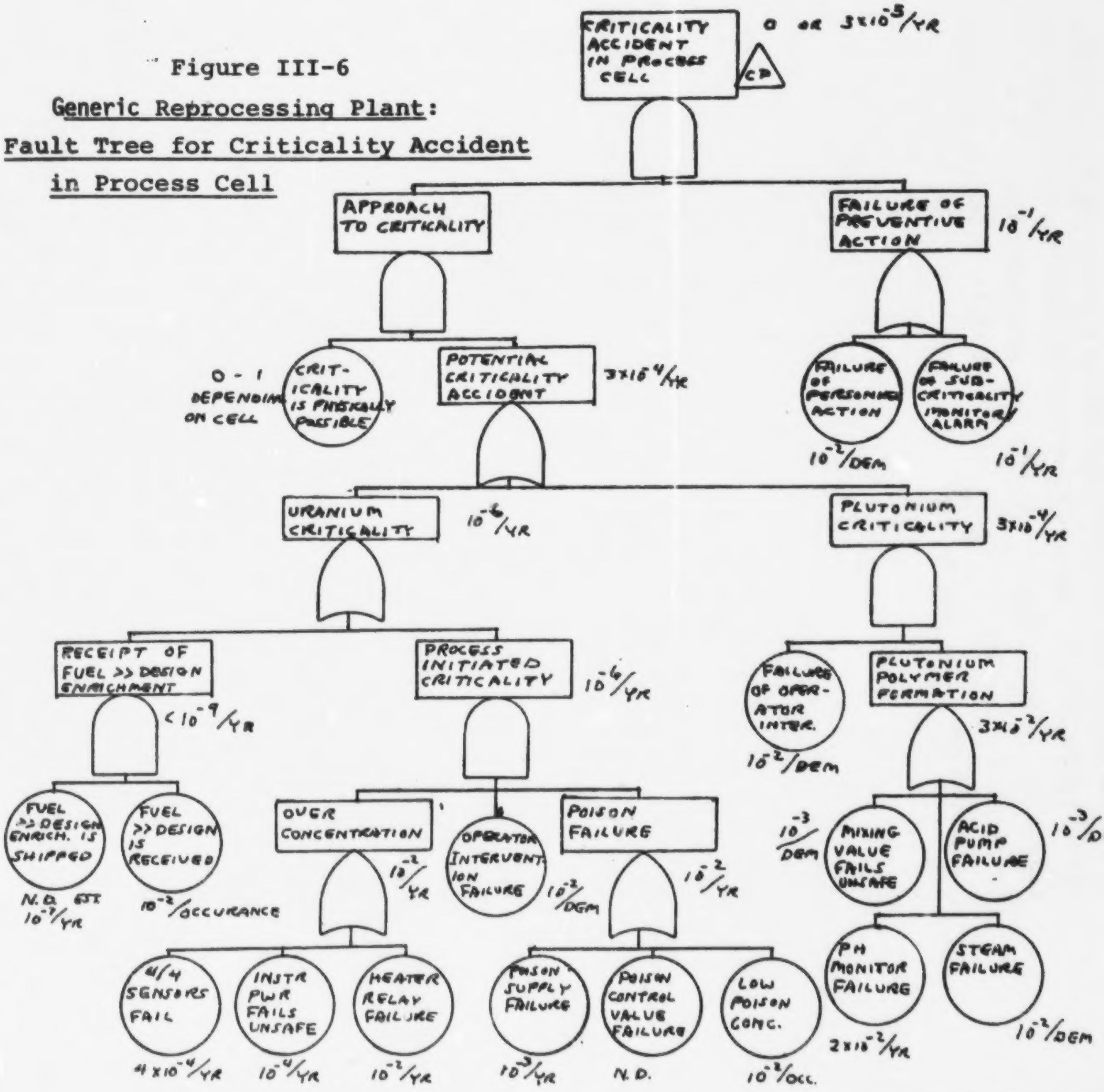
Figure III-5

Generic Reprocessing Plant:

Fault Tree for Red Oil Explosion

Figure III-6

Generic Reprocessing Plant:
Fault Tree for Criticality Accident
in Process Cell


CRITICALITY ACCIDENT IN PROCESS CELL
0 or 3×10^{-5}/YR
CP
APPROACH TO CRITICALITY
FAILURE OF PREVENTIVE ACTION
10^{-1}/YR
0 - 1 DEPENDING ON CELL
CRIT-ICALITY IS PHYSICALLY POSSIBLE
POTENTIAL CRITICALITY ACCIDENT
3×10^{-4}/YR
FAILURE OF PERSONNEL ACTION
10^{-2}/DEM
FAILURE OF SUB-CRITICALITY MONITOR ALARM
10^{-1}/YR
URANIUM CRITICALITY
10^{-6}/YR
PLUTONIUM CRITICALITY
3×10^{-4}/YR
RECEIPT OF FUEL >> DESIGN ENRICHMENT
$<10^{-9}$/YR
PROCESS INITIATED CRITICALITY
10^{-6}/YR
FAILURE OF OPER-ATOR INTER.
10^{-2}/DEM
PLUTONIUM POLYMER FORMATION
3×10^{-2}/YR
FUEL >> DESIGN ENRICH. IS SHIPPED
N.D. EST 10^{-7}/YR
FUEL >> DESIGN IS RECEIVED
10^{-2}/OCCURANCE
OVER CONCENTRATION
10^{-2}/YR
OPERATOR INTERVENT-ION FAILURE
10^{-2}/DEM
POISON FAILURE
10^{-2}/YR
10^{-3}/DEM
MIXING VALVE FAILS UNSAFE
ACID PUMP FAILURE
10^{-3}/DEM
4/4 SENSORS FAIL
4×10^{-4}/YR
INSTR PWR FAILS UNSAFE
10^{-4}/YR
HEATER RELAY FAILURE
10^{-2}/YR
POISON SUPPLY FAILURE
10^{-3}/YR
POISON CONTROL VALVE FAILURE
N.D.
LOW POISON CONC.
10^{-2}/OCC.
PH MONITOR FAILURE
2×10^{-2}/YR
STEAM FAILURE
10^{-2}/DEM

III-6 was taken from the Final Safety Analysis Report for the uninterruptable instrument power system in the Monticello power plant.[33] This also agrees with other data on d.c. system power.

This completes the failure probability analysis that began with figure III-3. The analysis continues by reference to figure III-2 and picking up another process cell. The complete collection of fault trees used in this report are presented in Appendix B.

IV. ACCIDENT ANALYSIS

1. General

A fuel reprocessing plant represents a substantial potential release source for radioactivity because of the presence of fissile material and fission products, the use of organic solvents and the handling of quantities of radioactivity in aqueous solutions. Because of the redundant safety systems designed into the plant and the inherent nature of the process, however, it is believed that most of the potential accidents have a low probability of occurrence.

Anticipated operational accidents are considered to be situations in which only one independent failure, human or equipment, is involved. Environmental radioactive releases from such events would be expected to be less than those allowable by 10 CFR 20 although the technical specifications on releases may be exceeded. An upper limit accident is considered to be a situation resulting from multiple operator errors, from multiple equipment malfunctions or from stresses imposed by natural phenomena which may have noticeable potential environmental consequences.

2. History of Accidents in Reprocessing

Most of the accidents that have occurred during irradiated fuel reprocessing operations took place at AEC installations when the industry was in its infancy. These experiences, in general, might not correspond with anticipated commercial plant experience. The Nuclear Fuel Services plant at West Valley, New York is the only commercial light water irradiated fuel reprocessing plant which has operated in this country. It had 6 years of operation before it suspended operation for modification to allow higher throughput[5]. Reprocessing experience that has been applied and operated on a production basis in the U.S. employing essentially the same process technologies, however, is greater than 100 plant-years[10].

A literature survey was made to accumulate a data base of accidents which were categorized into types of accidents for the nuclear energy and related chemical processing industries. The survey showed that they do happen in spite of safety precautions that have been taken. The data base provided guidance in the evaluation of the safety features assumed for the generic reprocessing plant too. Specific descriptions of the accident circumstances from the survey are presented in Appendix C.

The types of accidents which have occurred or are probable during reprocessing operations, according to the literature, are categorized as follows:

a) Criticality accidents
b) Fires
c) Explosions
d) Fuel receiving and storage accidents
e) Waste storage accidents
f) Natural phenomena events.

These are discussed in the following section.

3. Postulated Accidents; Bases for Selection in the Hazards Analysis

Accident situations selected for analysis were hypothesized from a review of reprocessing experiences, the design of the generic reprocessing plant and the unit operations involved in the process. The accidents selected are believed to have the most severe consequences in terms of potential exposure to the environment. These events, of low probability, are credible only if one assumes simultaneous failure of engineered safety features and where pertinent, compromise of administrative procedures established as safety barriers. The types of incidents considered are those most likely to result in the dispersion of radioactivity beyond the primary confinement. These include:

- Nuclear criticality - a solution criticality in a process vessel in the Remote Process Cell
- Fires involving solvent and process equipment; a fire in leached zirconium alloy cladding was also considered. The consequences of such an incident, however, were found to have a negligible potential environmental impact
- Explosions involving different types of radioactivity sources
- Fuel receiving and storage maloperation
- Loss of cooling to the high level waste facility
- Earthquake and tornado consequences on the model reprocessing plant's structures and systems containing the normal inventory of design basis fuel were also reviewed and evaluated.

The potential accidents are discussed in the following in terms of malfunctions or errors required for the accident to occur, the probabilities of their occurring, the protective measures available and the off-site consequences that could result from such accidental releases.

a. Criticality Accidents

In the 30 years of the nuclear age, representing 432 plant-years, there have been 30 criticality accidents in the U.S.[34,35]. Most of these occurred in experimental facilities; some of the early accidents are attributable to the limited understanding of nuclear energy at the time. There have been no criticality accidents since 1968. Seven of the accidents which occurred were in solution systems which could exist in a fuel reprocessing plant. The balance of the incidents involved conditions that would not be encountered in a reprocessing facility. A summary of the relevant accidents is presented in Table IV-1, page 61. The tabulated incidents occurred during reprocessing operations but not at a production facility. However, they provide anticipated general characteristics for criticality accidents which could occur in a processing plant's solution systems. These are summarized as follows:

1. The number of fissions in such an event would be less than 10^{20}
2. The accident would have to be sustained over many minutes to produce a fission yield approaching 10^{20} fissions

TABLE IV-1

Solution Criticality Accidents[34,35]

DATE	LOCATION	ACTIVE MATERIAL	GEOMETRY	TOTAL FISSIONS	CAUSE	PHYSICAL DAMAGE
June 16, 1953	Y-12 Processing Plant, Oak Ridge, Tenn.	2.5 kg ^{235}U $UO_2(NO_3)_2$ in 56 liters H_2O	Cylinder concrete reflected below	1.3×10^{18}	Wash water added to $UO_2(NO_3)_2$ solution	None
December 30,1958	LASL, New Mexico Pu Processing Plant	3.27 kg Pu $PuO_2(NO_3)_2$ in ∿ 168 liters H_2O	Cylinder water reflected below	1.5×10^{17}	Agitator created critical geometry	None
October 16,1959	Chemical Processing Plant Idaho Reactor	34.5 kg ^{235}U ∿800 liters $H_2O-UO_2(NO_3)_2$	Cylinder concrete reflected below	$\sim 4 \times 10^{19}$	Solution siphoned from safe to unsafe geometry	None
January 25,1961	Chemical Processing Plant Idaho Reactor Testing Area	8 kg ^{235}U $UO_2(NO_3)_2$ in 40 liters H_2O	Cylinder	6×10^{17}	Solution pumped from safe to unsafe geometry	None
April 7, 1962	Hanford Works, Richland, Wash.	1.55 kg Pu	Cylinder unreflected	8×10^{17}	Concentrated solution incorrectly siphoned	None
July 24, 1964	The Wood River Junction, R.I. scrap recovery facility	2.64 kg ^{235}U in $UO_2(NO_3)_2$	Cylinder unreflected	1.3×10^{17}	Concentrated solution poured into unsafe geometry. Additional moderation in tank	None
Jan. 30, 1968	Y-12 Processing Plant - Oak Ridge, Tenn.	3.3 Kg U-233 $UO_2(NO_3)_2$ in 20 ltr. water	Sphere water reflected	1.1×10^{16}	Solution surged from safe to unsafe geometry	None

3. The rate of energy release would be too low to be explosive, i.e., no shock front generation would be anticipated

4. The event would be associated with a change from normal procedures

5. The environmental impact would be very small; total property loss would be less than $70,000 as indicated from the incidents which occurred.

Criticality incidents have typically resulted in initial bursts of less than 10^{18} fissions followed in some instances by subsequent bursts of less than 3×10^{16} fissions per second. Little or no damage resulted to the confinement equipment from the criticality events.

In this analysis, a criticality is assumed to occur in the Remote Process Cell. The assumptions concerning the radioactive release are discussed in section V on consequences. Accidental criticality in fuel receiving and storage operations is unlikely because the areas where these operations are performed are designed to be subcritical with unirradiated fuel of 5% enrichment. Light water reactor fuel is normally enriched to less than 4%. After a burnup of 30,000 MWD/tonne the enrichment is reduced to less than 3%. Fission products generated in the irradiation also contribute neutron poisons in the elements thereby further reducing their fuel worth.

Criticality in the Remote Process Cell could accidentally occur by overfilling a dissolver through a failure to switch the diverter chute. Depending on geometry, this would have to occur in conjunction with a reduction in neutron poison and failure of monitoring instruments.

In the multi-batch dissolution process if the filter (40 mesh) fails or is eliminated it is possible to transfer plutonium fines to the dissolver transfer tank and to the accountability tank. Accidental transfer of fines to the HAF feed tank is minimized or eliminated by centrifugal clarification of the feed after it leaves the accountability tank.

Some possibility of criticality exists in the centrifuge. This is made unlikely by an interlock on excessive current to prevent further operation if it is loaded with more than 1.5 kg of insoluble fines.

Following centrifugal contacting and concentration of uranium and plutonium, criticality is inhibited by controlling the concentrations of fissile material in the solutions. Criticality could result from the formation of an excessive fissile material concentration due to out-of-specification process control coincident with the failure of the concentration monitors. As an example, the dilute aqueous plutonium stream (1BP) is continuously monitored for plutonium concentration. The density is also checked by the analysis of grab samples. In addition to these monitors, neutron monitors are mounted on the HS column (scrubber) and the 1B electropulse column (product separation) to warn of excessive plutonium concentration.

A criticality accident could occur in this part of the process from a failure of process control that results in higher-than-normal fissile material concentrations in solution concurrently with multiple monitoring failures or it could result from administrative error by processing higher enrichment fuel under specifications normally used for lower uranium enrichment.

Criticality in product loadout is also possible. It would be of higher consequence in the plutonium loadout area than in the uranium loadout area. Criticality is avoided by density control, neutron monitoring, use of equipment having favorable geometry, and fixed neutron poisons. The formation of plutonium polymers is controlled by keeping the solution greater than 1 M acid. The plutonium product is stored in product tanks of favorable geometry designed to earthquake and tornado criteria both as to integrity and position.

Criticality in product loadout could result from failure of concentration control coincident with monitor failure. It could result from plutonium polymer formation coincident with monitor and acid control failure. It could result from flooding of the plutonium product cell or greater than design basis earthquake or tornado.

b. Fires

Three postulated incidents involving dispersal of radioactive material through the agency of fire have been analyzed. One involves a contaminated solvent, assumed to contain a substantial loading of iodine along with plutonium and fission products. Another involves a solvent fire in the plutonium extraction cycle. The third fire evaluated is assumed to occur with the ion-exchange resin during product purification.

A fire involving leached hulls (chopped cladding containing residual fuel material) was initially considered too. Experience in six years of shearing and handling the leached zirconium hulls indicated a very low probability of a major fire in this material[5]. More than 95% of the radioactivity associated with the leached hulls is induced radioactivity and is an integral part of the metal itself. In order for this radioactivity to escape, the metal itself must burn, potentially producing volatile particulates. The remainder of the radioactivity associated with the hulls is fission product and transuranic species in the form of refractory oxides. All of this radioactivity which remains with the hulls failed to be removed from the hulls during several hours of vigorous boiling in nitric acid and subsequent washing. In order for a hull fire to occur, there would have to be source of ignition of sufficient energy to initiate an active fire in the zirconium hulls which would either go undiscovered, or not respond to firefighting efforts using dry chemical extinguishing agents. There have been instances under which leached zirconium hulls have glowed but at no time has this ever affected more than a small fraction of the hulls nor has it ever resulted in anything more active than a transient glowing of the few hulls involved. The probability of such an event going unnoticed is small also, as there is an operator present at the viewing window of the hull inspection and canning station whenever they are being handled. In comparison to the hypothetical solvent fire, a zirconium fire results in lower potential off-site releases and the heat release in such an incident is estimated to be approximately 5% that for the postulated solvent fire[36].

1. Solvent Fires

Codecontamination is the operation which removes most of the fission products and other undesirable impurities from solutions of uranium and plutonium without separating the uranium and plutonium components. Partitioning is the operation in which the uranium is separated from the plutonium (neptunium). Solvent extraction cycles, employing 30 volume percent tributyl phosphate (TBP) in a normal paraffinic hydrocarbon (dodecane) as the water immiscible solvent, is used for these operations in conjunction with various chemical adjustments. Because of the moderate flash point of the organic solvent, 70°C, there is a potential for the occurrence of solvent fires during these processing steps due to upsets or system leaks. Operating temperatures are held below the 70°C flash point by temperature controls and flow rates are monitored to avoid spills and to maintain the desired compositions in all feed and discharge streams of the equipment used in these processing steps. This applies to the use of anion exchange resin columns or electrochemical reductions for the partitioning steps.

A solvent fire could result from a failure of temperature control which would allow the flash point to be reached. Loss of temperature control could be caused by failure of the temperature sensor, temperature control servo or mixing valve failure. Leaks or spills due to process control upsets or pipe rupture under earthquake conditions could result in a solvent fire if failure of sump level sensors to solvent accumulations on the cell floors occurred.

2. Ion Exchange Resin Fires

Ion exchange resin columns are used to partition plutonium, uranium and/or neptunium into separate streams while providing for additional fission product decontamination following the initial codecontamination step. Potential autothermal resin-nitrate reaction in these ion exchange columns can be prevented in this processing step by limiting the temperature to less than 135°C and by avoiding contact of the material with oxidizing agents.

A resin fire could result from failure of process control to limit the acidity as monitored by specific gravity measurement of the feed streams. Monitoring is done continuously and alarmed on high density indication. Failure of such alarms and monitors to indicate a potentially hazardous condition could lead to a resin fire.

Failure of temperature control to maintain the resin temperature to less than 135°C could also result in a resin-nitrate reaction. Temperature regulation is performed by sensors with servo controls. Thus, anything that can upset this control, such as failure of the sensor or the servomechanism, could lead to excessive heating and an incident.

Low pressure steam is commonly used for heating purposes in reprocessing plants. Under upset conditions such that the process came into temperature equilibrium with the steam, a temperature of 135°C could be attained which could lead to the onset of a resin-nitrate reaction.

If the resin beds were highly loaded with plutonium, radioactive heating could augment the temperature of the bed. This could occur in the plutonium purification and recovery operation carried out in the plutonium product cell. Failure to remove the product from loaded resins in the event of a plant shutdown would also be conducive to initiating such an incident.

Mitigating safeguards, i.e., the use of heavy bar grids to inhibit the expulsion of resin from the columns and pressure relief instrumentation, are incorporated in the equipment design to minimize the consequences of such an event.

Advancing reprocessing technology may preclude the need for utilizing ion exchange resins in separations operations. The occurrence of such an accident was therefore excluded from consideration in this analysis.

c. Explosions

Postulated incidents involving the dispersion of radioactivity by explosions in process equipment are analyzed. Explosions considered included the following:

- an explosion in the high aqueous feed tank (HAF)
- an explosion in the high aqueous waste concentrator (HAW concentrator)
- an explosion in the low aqueous waste concentrator (LAW concentrator)
- an explosion in the silver zeolite iodine adsorber
- an explosion in the waste calciner
- an explosion in the plutonium product calciner.

An explosion in the high aqueous feed tank could conceivably be caused by ignition of an explosive mixture of radiolytically generated hydrogen in the air above the liquid.

An explosion in the HAW or LAW concentrator could conceivably be caused by ignition of an explosive mixture of radiolytically generated hydrogen in the air above the liquid in the evaporator or a "red-oil" explosion. An explosion in the silver zeolite iodine adsorber could result from the formation of silver azide due to the presence of excess ammoniacal material in the off-gases. A waste calciner explosion could also conceivably occur from the excessive presence of hydrogen and/or "red-oil" in the equipment used in this unit operation.

An explosion in the plutonium product calciner could conceivably result from ignition of an explosive mixture of hydrogen which may be used in the process of decomposing the intermediate plutonium oxalate product or from an accelerated decomposition of moist oxalate crystalline material.

These circumstances are normally precluded from developing by means of both design and operational safety features; multiple failures of these protective systems could conceivably lead to the incidents noted.

"Red-oil" is a material that can be formed from a heavy metal nitrate, e.g., uranium, and/or nitric acid solutions mixed with tributyl phosphate solvent at temperatures exceeding 135°C[37,38]. The exact nature of the reaction of tributyl phosphate (TBP) with hot concentrated solutions of heavy metal nitrates and nitric acid has not been determined. However, under optimum conditions, the reaction becomes explosive and oxides of nitrogen are evolved. Although Purex-process conditions do not approach those giving rise to such a reaction, safeguards are provided in the concentration of uranium, plutonium and nitric acid solutions to prevent their accidental occurrence. One method of avoiding an explosion is to keep the evaporation temperature below 135°C, the minimum temperature at which the reaction occurs. Another method is to remove the TBP from the aqueous stream prior to evaporation by steam stripping the aqueous stream.

In order for a "red-oil" explosion to occur, several independent instrument control failures and administrative failures would have to precede the occurrence. The code-contamination column would have to be out of control dumping solvent with the high aqueous waste (HAW) stream for a number of hours without being noticed or corrected; the controller for the reduction of 150 psig steam (181°C) to less than 40 psig steam (131°C) would have to malfunction; the relief valve which restricts the low pressure steam would have to fail to operate; the pressure controller on the steam to the evaporator heating coils would have to malfunction, causing steam pressure to rise above 40 psig, and not be noticed for an extended period of time, and the evaporator bottoms product would have to be substantially overconcentrated while all the normal indicators of this condition either malfunction or are ignored. Given these conditions, an explosion involving complexes of a heavy metal, TBP and nitric acid is possible.

Radiolysis of aqueous solutions results in the production of hydrogen and oxygen. Thus, all process streams would be expected to evolve hydrogen. It is estimated that 3 ft^3 of hydrogen (STP) per 10^6 BTU of fission product heat in acid waste and 10 $ft^3/10^6$ BTU in alkaline waste is formed by radiolysis[11]. The combustion threshold of hydrogen in air occurs at 4% concentration. To avoid reaching this hydrogen concentration, dilution of the off-gases with continuously flowing air is used in the processing operations. The HAF storage tank, the HAW and LAW concentrators and the high and intermediate level liquid waste systems have the highest potential for such an explosion due to hydrogen gas accumulation as a result of failure of the air purge system. Such a failure would constitute a common mode failure to all the systems.

To reduce the likelihood of air flow failure, the plant is designed with redundant air flow features which have been discussed under special safety systems. These systems include redundant ventilation blowers with a spare blower in reserve, emergency electrical power, air reservoirs and spare air compressors.

A hydrogen explosion might occur as a result of failures in the process ventilation system such as two DOG/VOG blowers failing together or their power failing, filter blockages, ventilation control failure such that air flow balancing dampers are closed and failure of emergency compressed air which could result in a hydrogen explosion in the HLLW or ILLW storage tanks if the waste is not stored under self-boiling conditions.

The process off-gas system, composed of the Dissolver Off-Gas (DOG) and Vessel Off-Gas (VOG) systems removes iodine, particulate radioactive contaminants and nitrogen oxides which are volatilized during the fuel segment dissolution and subsequent processing steps. These off-gases are then combined with the ventilation air, filtered through two stages of HEPA off-gas filters and discharged through the stack.

Silver zeolite sorbents are incorporated in the subsystem scrubbing trains to further restrict volatile radioiodine releases to the environment.(39)

An explosion in the waste calciner facility can be postulated on essentially the same basis as the high level waste concentrator explosion for the calciner would be fed from the HAW concentrator. The calciner, however, operates at a much higher temperature than the concentrator (>450°C). This would tend to increase the quantity of ruthenium that could be volatilized during an accident by about a factor of 10 over the amount that could be volatilized from a HAW concentrator explosion[40-43]. Aside from this variation and the fact that the equipment employed in this operation, having a small holdup which would limit the probability of such an incident, the results of a waste calciner explosion should be essentially the same as for the HAW concentrator explosion analysis.

An explosion in the plutonium product concentrator can be postulated on essentially the same basis as the high level waste concentrator explosion. The hydrogen generation rate from a plutonium product solution, however, is much lower than from a high level waste solution. Hoover and Ingalls[44] quote a hydrogen generation rate for plutonium as 0.05-0.06 ml/day/gram Pu-239 at the probable nitric acid concentration range anticipated in the evaporator concentrate. Increasing this generation rate by an order of magnitude, to reflect the higher specific activity of design basis plutonium, and assuming an equilibrium quantity of process solution of 10 liters at 200 grams per liter of plutonium, the rate of hydrogen generation in the evaporator would be about 50 ml per hour. At a volume free space in the evaporator of 38 liters, at least 30 hours would be required to reach the minimum lower explosive limit of hydrogen content, disregarding the flow of instrument air to the probes. (This in itself would be sufficient to keep all hydrogen generated at a concentration below the lower explosive limit for the equipment has a small holdup capacity.)

Although the plutonium product concentration explosion is highly improbable, it has been evaluated as an upper limit accident for the inventory of the plutonium in the concentrator is large - 2,000 grams or 23,000 total curies at the time of the postulated explosion[36]. Using the same assumptions as were employed to analyze the high activity waste evaporator explosion, the radioactivity release is calculated to be 1.5×10^{-2} curies.

A criticality and/or explosion accident during conversion of plutonium nitrate to oxide powder is also considered to be highly improbable. In this operation, plutonium nitrate in solution from anion exchange or evaporation is precipitated as the oxalate. This product is filtered and washed. The wet oxalate crystals are dried at 400°C for a fixed period and then calcined at 750°C also for a fixed period, in a slow stream of air or hydrogen. To ensure criticality control, fissile concentration controls and favorable geometry with fixed poisons are employed in the operations. Batch sizes of solids processed are always maintained subcritical. The product powder is screened, sampled, weighed and sealed in metal containers for subsequent shipment or storage.

The governing radioactive material that could be released to the environment due to such an accident would be in a particulate form. Decomposition of plutonium oxalate to oxide results in particles having an indicated Mass Median Diameter of 8-12 microns[45]. Oxide particles were found to be 26 to 68% of the size of their precursor. These measurements were made under laboratory conditions; plant oxide produced under accident conditions will probably be coarser. Mishima et al, report fractional releases up to 1% of the source when heating either the oxalate or partially oxidized oxalate in an upsweep of air[46]. Their finely divided, free flowing powder was composed of spheres with a Mass Median Diameter of 32 microns. As much as 0.9% of the plutonium used in the source was made airborne during a 1-hour period at temperatures up to 1000°C and air velocities up to 100 cm/sec (∿2.5mph).

For the postulated accident, it is assumed that in the startup of the calcining furnace, hydrogen is introduced before the air is displaced with nitrogen, violating the procedure and resulting in a hydrogen-air explosion. The oxalate batch size being calcined is assumed to contain the equivalent of 3 kg of Pu. Overpressurization of the furnace will not be enough to destroy the furnace. Consequently, the explosion would be directed toward the ends of the furnace. The powder is assumed to be entrained at a windspeed of 20 mph. At this condition, up to 15% of the solids would be expected to be airborne into the ventilation exhaust system. The airborne powder particles' size is so large that the filtering efficiency of the three stages of HEPA filters in series would be almost 100%. A filter release factor of 8×10^{-9} is assumed[20].

If the furnace does explode, at most the surrounding glove box might rupture and powder would be ejected into the room. This would create a temporary internal contamination problem in the building. However, the size of the particles would be too large to cause a sizeable external release of radioactivity. It is assumed a 1% release to the exhaust ventilation system could occur. The anticipated stack releases for the cases discussed are as follows for a Pu mixture having a specific activity of 14.1 Ci/g[19].

1. Furnace intact, direct release to vent system:

   $(3000 \text{ g Pu})(0.15)(8 \times 10^{-9}) = 3.6 \times 10^{-6} \text{g Pu} \approx 5.1 \times 10^{-5} \text{ Ci}$

2. Furnace explodes with release to room:

   $(3000\text{g Pu})(0.01)(8 \times 10^{-9}) = 2.4 \times 10^{-7} \text{g Pu} \approx 3.4 \times 10^{-6} \text{ Ci}$

The off-site release from such incidents will have insignificant environmental impact even under the worst dispersion conditions for the HEPA filters will reduce these values by a factor of $>10^{+3}$.

d. Fuel Receiving and Storage Area Accident

An accident in the fuel receiving and storage area resulting in the release of radioactivity that could have an environmental impact is a credible event. The consequences of an uncontrolled release in this plant area could be serious although the probability of such an occurrence would be small. Consideration of such an event was made in this analysis.

Based on regulatory standards and requirements for package design, quality assurance and handling and monitoring procedures, the probability of a cask being breached is low[47].

A hypothetical incident which may bound a variety of events as to the nature and the magnitude of a release in the fuel receiving and storage area is assumed for this evaluation and is described as follows:

> In shipment, it is assumed that the spent fuel cask loses its heat removal capability. The spent fuel rods self-heat due to fission product decay heating to a temperature approaching 1225°C. This causes cladding failure and release of a large fraction of the more volatile fission products to the hermetically sealed, dry cask cavity. After receipt of the shipping cask at the reprocessing plant, it is transferred to the cask unloading pool in the fuel receiving and storage area where it is opened. On opening the cask, mobile radioactive species are expelled from the cask cavity as a stream of bubbles which rise to the pool surface. Some of the fission products released in the cask cavity will plate out on the cask's internal surface; some will remain in the pool water. The balance of the fission products - primarily volatile species - will be airborne within the building area and are assumed to pass through the vessel off-gas system. Of the fission products in the spent fuel cask inventory, it is assumed that

all of the noble gases, tritium, halogens, cesium, strontium and ruthenium in the breached elements are released to the unloading pool.[48,49] For this evaluation, the airborne release of the noble gases and tritium is neglected since they are accepted as normal releases to the atmosphere. Ninety percent of the halogens (essentially iodine), ruthenium and cesium and all of the strontium expelled from the cask are conservatively assumed to remain in the pool water. The balance are airborne and pass through the vessel off-gas system.

Table IV-3 lists the anticipated release data for the radioactive species involved.

e. Leakage of Fission Products From High Level Waste Tanks

High level wastes will be stored on an interim basis in high integrity tanks as solutions, with the radiolytic heat removed by heat transfer to a cooling water system. As a safety device for the eventuality of failure of the cooling system water supply, provision is included for venting the storage tanks to the atmosphere via an off-gas system designed for a total decontamination factor, boiling waste to atmosphere, of at least 1×10^{10}.[36,40,41] Because of the defenses in depth which would be operative - high integrity design of the cooling water supply system, tank-saucer-vault containment design, a failure and/or

TABLE IV-3(50,51)

Anticipated Releases - Fuel Receiving And Storage Area Incident

Group	Isotope:	Ci/Tonne	Source/4.5 tonne Shipment	Fraction Released
Ru:				
	Ru-103	1.2x10^{5}	5.4(10^{5})	10%
	Ru-106	6.1x10^{5}	27.4(10^{5})	
I:				
	I-129	3.6x10^{-2}	16.2(10^{-2})	10%
	I-131	1.6	7.2	
	Cs-134	2.4x10^{5}	10.8(10^{5})	10%
OFP*	Cs-137	1.2x10^{5}	5.4(10^{5})	

* Other Fission Products

accident in this part of the facility which could have an environmental impact is not expected. Tank storage of high level wastes has been accomplished safely over the 25 years since they began to be generated, despite the fact that some leaks have occurred (see Appendix C). Extensive measurements at the leak locations show that the material released has remained in the vicinity of the tanks. It is anticipated that developing technology will incorporate additional capability to detect and contain any leakage that might occur.

f. Natural Phenomena Incidents

The reprocessing plant is designed, evaluated and constructed to criteria and guidelines accepted as adequate to provide reasonable assurance that the plant could be operated without undue risk to the health and safety of the public[53]. These criteria include requirements that the structures, systems and components important to safety, be designed to withstand the effects of natural phenomena (earthquakes and tornadoes). The design bases that satisfy the natural phenomena criteria reflect:

- Appropriate consideration of the most severe of the natural phenomena that has historically been reported for the site and the surrounding area
- Appropriate combinations of the effects of normal and accident conditions with the effects of natural phenomena
- The importance of the safety functions performed.

Because (1) operating and accident stresses are lower, (2) process parameters cannot change as rapidly, and (3) consequences of any failure are generally less severe for fuel reprocessing operations, no plant components have performance requirements that are critical to safety assurance in the same sense as power reactor safety system response or cooling continuity requirements. Protection against uncontrolled release of radioactive materials is assured by maintaining the mechanical and structural integrity of relatively passive confinement and off-gas treatment facilities and of certain monitoring and control system components. No violent expulsion of process materials would be anticipated in the event of a natural phenomenon incident as system components are not highly stressed during operation. Operating loads are accounted for in design and construction because system components and interconnections are generally of small size (over 90% of the process piping is less than 2 inches in diameter) and relatively thick - walled as a result of corrosion allowances provided. The availability of multiple confinement barriers further assures that radioactive materials escaping from the primary process systems or from effluent collection, treatment or disposal facilities are monitored and controlled so as not to result in uncontrolled releases to the environment[5-7].

An earthquake may negate some or all of the confinement of individual systems but will have little, if any, direct dispersive mechanisms.

The potential off-site exposures from such an event would be much less than one percent of the accident exposure guideline[36].

The plant structure will also protect the radioactive inventory from the effect of a design basis tornado. The Class 1 Ventilation System which ventilates the cells and the emergency cooling system for potentially self-boiling solutions will maintain the process in a safe shutdown condition. The inventories of the plant areas which have a potential for release during or subsequent to a design basis tornado would be negligible. Although the tornado has great potential for releasing radioactivity, it has great potential for dispersing the release thereby mitigating the consequences. The maximum potential off-site releases and exposures would be well within the guidelines[36].

V. Consequences of Accidents

For the purpose of demonstrating the degree of inherent safety of the generic reprocessing plant and its confinement systems, the credible consequences of operational upsets and of stresses which might be imposed by the design basis natural phenomena were selected in evaluating the postulated accidents. The accidents examined are those believed to have the most severe consequences in terms of potential exposure to the environment. These accidents are events of low probability which are credible only if one assumes simultaneous failure of engineered safety features and where pertinent, administrative procedures established as safety barriers are bypassed.

For each accident probability sequence, there is, at that point in the reprocessing operations, a corresponding consequence of radioactivity dispersal beyond the primary confinement. These source term release values have been calculated for generic reprocessing plant accidents using realistic values for many of the fault tree branches discussed in Section III and Appendix B.

The types of accidents considered were discussed in Section IV. The cases examined are listed in Table V-1. Some of these accidents could appear in more than one location on a fault tree or on several fault trees. The physical assumptions used in the analyses are generally based upon the design of the Barnwell Nuclear Fuel Plant.[7]

TABLE V-1

Postulated Reprocessing Plant Accidents Examined

HAW Concentrator Explosion
Solvent Fire in the Plutonium Extraction Cycle
Solvent Fire in the Co-Decontamination Cycle*
Explosion in the LAW Concentrator
Ion Exchange Resin Fire*
Nuclear Criticality Incident
Explosion in the HAF Tank
Waste Calciner Explosion
Fuel Receiving and Storage Accident

*Accident Examined Was In The Neptunium Recovery Cycle

Accident releases might be initiated in any of the number of process cells in the reprocessing plant. Airborne releases would obviously have to pass through the plant's ventilation system prior to escaping to the environment. In the realistic calculations the following measured filter efficiencies were used in developing the source terms:[19,20]

First HEPA filter removes	99.9%
Second HEPA filter removes	99.0%
Third HEPA filter removes	94.0%
Fourth HEPA filter removes	83.0%

The accidents listed in Table V-1 are considered in detail in the following:

1. HAW Concentrator Explosion

The assumptions and results for this accident as calculated in the Barnwell SAR[7] are given in Table V-2.

TABLE V-2

SAR Source Calculation HAW Concentrator Explosion (Barnwell)

Group*	Isotope	Ci/Tonne	x SF	x RF	x EF	= Ci Released
Ru:			1.0	1.0×10^{-3}	1.05	
	Ru-103	1.2×10^{5}				126
	Ru-106	6.1×10^{5}				640
Zr-Nb:			1.0	4.6×10^{-7}	1.05	
	Zr-95	3.5×10^{5}				.17
	Nb-95	6.5×10^{5}				.31
I:			8.3×10^{-2}	1.0	1.05	
	I-129	3.6×10^{-2}				3.2×10^{-3}
	I-131	1.6				0.14
OFPT:			1.0	4.6×10^{-7}	1.05	
	Sr-89	9.0×10^{4}				.043
	Sr-90	8.4×10^{4}				.042
	Y-90	8.4×10^{4}				.042
	Y-91	1.9×10^{5}				.094
	Cs-134	2.4×10^{5}				.119
	Cs-137	1.2×10^{5}				.060
	Ce-141	7.9×10^{4}				.039
	Ce-144	8.8×10^{5}				.434
	Pm-147	1.4×10^{5}				.070
	Am-241	250.				1.2×10^{-4}
	Am-242	4.0				2.1×10^{-6}
	Cm-242	4.4×10^{4}				.021
	Cm-243	34.0				1.7×10^{-5}
	Cm-244	5.7×10^{3}				.0028
Pu:			0.01	4.6×10^{-7}	1.05	
	Pu-238	4.3×10^{3}				2.0×10^{-5}
	Pu-239	3.2×10^{2}				1.5×10^{-6}
	Pu-240	6.3×10^{2}				2.9×10^{-6}
	Pu-241	1.7×10^{5}				8.3×10^{-4}
	Pu-242	3.6				1.7×10^{-8}

Release Factor:

Non-volatile fraction dispersed:

Concentrator volume = 600 liters of solution.
Cell volume = 2850 m^3; dispersion = 100 mg/m^3
Total solution dispersed = 0.285 kg of concentrate
Density of concentrate = 1.4 kg/ℓ
Fraction of concentrate dispersed = 3.3×10^{-4}
Dispersion passing through filter = 0.14 mg/m^3
Filter factor = 1.4×10^{-3}

Element	Non-Volatile Fraction Dispersed	x	Filter Factor	+	Volatile Fraction	=	RF
Ru	3.3×10^{-4}		1.4×10^{-3}		1×10^{-3}		1.0×10^{-3}
Zr-Nb	3.3×10^{-4}		1.4×10^{-3}		0		4.6×10^{-7}
I	0				1.0		1.0
OFPT	3.3×10^{-4}		1.4×10^{-3}		0		4.6×10^{-7}
Pu	3.3×10^{-4}		1.4×10^{-3}		0		4.6×10^{-7}

Equivalence Factor:

EF = 600 liters of concentrate/567ℓ/MTU = 1.05 MTU

Since the HAW Concentrator is in the Remote Process Cell (RPC), there are 2 HEPA filters in series before stack release. To carry out a realistic calculation one needs the non-volatile release fraction which is (2850 m^3 cell volume)/(100mg/m^3), the denominator being a representative density.[3] This yields a release factor RF = 3.33×10^{-4} for non-volatiles. In addition 100% of the iodine and 0.1% of the ruthenium are volatilized. With these data the following releases are calculated.

TABLE V-3

Volatile Source From HAW Concentrator Explosion

Gro	Ci/tonne	X	SF	X	EF	+	Volatile Fraction	Volatile Sources Ci
RU-103	1.2E5		1		1.05		1.0E-3	1.26E2
Ru-106	6.1E5		1		1.05		1.0E-3	6.41E2
I-129	3.6E-2		8.3E-2		1.05		1.0	3.14E-3
I-131	1.6E0		8.3E-2		1.05		1.0	1.39E-1

* Note: 1E5 = 1×10^5

Table V-4

Non-Volatile Source From HAW Concentrator Explosion

Group	Isotope	Ci/Tonne	X	SF	X	EF	X	RF	No Filter Working	One Filter Working	Two Filters Working
Ru:				1.0		1.05		3.3E-4			
	Ru-103	1.2E5							4.16E1	4.16E-2	4.16E-4
	Ru-106	6.1E5							2.12E2	2.12E-1	2.12E-3
Zr-Nb				1.0		1.05		3.3E-4			
	Zr-95	3.5E5							1.21E2	1.21E-1	1.21E-3
	Nb-95	6.5E5							2.25E2	2.25E-1	2.25E-3
I:				8.3E-2		1.05		0			
	I-129	3.6E-2							0	0	0
	I-131	1.6E0							0	0	0
OFFT:				1.0		1.05		3.3E-4			
	Sr-89	9.0E4							3.12E1	3.12E-2	3.12E-4
	Sr-90	8.4E4							2.91E1	2.91E-2	2.91E-4
	Y-90	8.4E4							2.91E1	2.91E-2	2.91E-4
	Y-91	1.9E5							6.63E1	6.63E-2	6.63E-4
	Cs-134	2.4E5							8.33E1	8.33E-2	8.33E-4
	Cs-137	1.2E5							4.17E1	4.17E-2	4.17E-4
	Ce-141	7.9E4							2.74E1	2.74E-2	2.74E-4
	Ce-144	8.8E5							3.05E2	3.05E-1	3.05E-3
	Pm-147	1.4E5							4.81E1	4.81E-2	4.81E-4
	Am-241	250.0E0							8.67E-2	8.67E-5	8.67E-7
	Am-242	4.0E0							1.39E-3	1.39E-6	1.39E-8
	Cm-242	4.4E4							1.52E1	1.52E-2	1.52E-4
	Cm-243	34.0E0							1.18E-2	1.18E-5	1.18E-7
	Cm-244	5.7E3							1.98E-0	1.98E-3	1.98E-5
Pu:				0.01		1.05		3.3E-4			
	Pu-238	4.3E3							1.49E0	1.49E-3	1.49E-5
	Pu-239	3.2E2							1.11E-1	1.11E-4	1.11E-6
	Pu-240	6.3E2							2.18E-1	2.18E-4	2.18E-6
	Pu-241	1.7E5							5.9E1	5.9 E-2	5.9 E-4
	Pu-242	3.6E0							1.25E-3	1.25E-6	1.25E-8

* Note: 1.2E5 = 1 x 10^5

2. Solvent Fire in Pu Extraction Cycle

The Barnwell SAR[7] calculation is given below.

TABLE V-5

SOLVENT FIRE IN THE Pu SOLVENT EXTRACTION CYCLE

Group	Isotope:*	Ci/Tonne	X	SF	X	RF	X	EF	=	Ci Release
Pu:										
	Pu-238	4.3×10^{3}		1.0		1×10^{-4}		0.034		0.015
	Pu-239	3.2×10^{2}		1.0		1×10^{-4}		0.034		0.0011
	Pu-240	6.3×10^{2}		1.0		1×10^{-4}		0.034		0.0021
	Pu-241	1.7×10^{5}		1.0		1×10^{-4}		0.034		0.58
	Pu-242	3.6		1.0		1×10^{-4}		0.034		$1.3\times10^{-}$

* Other radioisotopes are considered to be negligible relative to plutonium.

Equivalence Factor:

Organic consumed = 14 liters
Processing rate = 0.208 MTU/hr
3AP normal flow = 85 ℓ/hr = 409 ℓ/MTU

$$EF = \frac{14\ \ell}{409\ \ell/MTU} = 0.034\ MTU$$

Release Factor:

Pu dispersed = 1%
Filter release = 1%
RF = (.01)(.01) = 1×10^{-4}

This incident would occur in the Plutonium Product Cell (PPC). There are 3 HEPA filters in series before stack release. The calculation yielded the following data:

TABLE V-6

Solvent Fire in the Pu Extraction Cycle

								Stack Release Curies			
Group	Isotope:*	Ci/Tonne	X	SF	X	EF	Fraction Dispersed	No Filter Working	One Filter Working	Two Filters Working	Three Filters Working
Pu:											
	Pu-238	4.3×10^3		1.0		0.034	.01	1.46E0*	1.46E-3	1.46E-5	8.76E-7
	Pu-239	3.2×10^2		1.0		0.034	.01	1.09E-1	1.09E-4	1.09E-6	6.54E-8
	Pu-240	6.3×10^2		1.0		0.034	.01	2.14E-1	2.14E-4	2.14E-6	1.28E-7
	Pu-241	1.7×10^5		1.0		0.034	.01	5.77E1	5.77E-2	5.77E-4	3.46E-5
	Pu-242	3.6		1.0		0.034	.01	1.22E-3	1.22E-6	1.22E-8	7.32E-10

* 1.46E0 = 1.46×10^0

(Neptunium Recovery Campaign)

The Barnwell[7] Calculation is given in Table V-7.

TABLE V-7 - SAR Results

Group*	Isotope	Ci/Tonne	x	DF**	x	SF	x	RF	x	EF	=	Ci Released
Ru:						0.01		0.10		0.085		
	^{103}Ru	1.2×10^5		0.16								1.69
	^{106}Ru	6.1×10^5		0.73								38
Zr-Nb:						0.01		1×10^{-4}		0.085		
	^{95}Zr	3.5×10^5		0.14								4.1×10^{-3}
	^{95}Nb	6.5×10^5		0.14								7.7×10^{-3}
I:						0.04		1.0		0.085		
	^{129}I	3.6×10^{-2}		1.0								1.2×10^{-4}
	^{131}I	1.6		0.032								1.8×10^{-4}
OFPT:						0.001		1×10^{-4}		0.085		
	^{89}Sr	9.0×10^4		0.20								1.5×10^{-4}
	^{90}Sr	8.4×10^4		1.0								7.2×10^{-4}
	^{90}Y	8.4×10^4		1.0								7.2×10^{-4}
	^{91}Y	1.9×10^5		0.23								3.6×10^{-4}
	^{134}Cs	2.4×10^5		0.79								1.7×10^{-3}
	^{137}Cs	1.2×10^5		0.96								1.0×10^{-3}
	^{141}Ce	7.9×10^4		0.13								8.4×10^{-5}
	^{144}Ce	8.8×10^5		0.66								5.0×10^{-3}
	^{147}Pm	1.4×10^5		0.90								1.1×10^{-3}
	^{241}Am	250.		1.0								2.1×10^{-6}
	^{243}Am	4.0		1.0								3.4×10^{-8}
	^{242}Cm	4.4×10^4		0.43								1.6×10^{-4}
	^{243}Cm	34.0		0.99								2.9×10^{-7}
	^{244}Cm	5.7×10^3		1.0								4.8×10^{-5}
Pu:						0.01		1×10^{-4}		0.085		
	^{238}Pu	4.3×10^3		1.0								3.6×10^{-4}
	^{239}Pu	3.2×10^2		1.0								2.7×10^{-5}
	^{240}Pu	6.3×10^2		1.0								5.4×10^{-5}
	^{241}Pu	1.7×10^5		1.0								1.5×10^{-5}
	^{242}Pu	3.6		1.0								2.7×10^{-7}

*OFPT = Other fission products and transuranium elements.

**DF = Decay Factor; corrects for simultaneous accumulation and decay of stored waste for one year before Np recovery campaign.

Release Factor:

99% filter efficiency.
1% of contaminants in burned organic dispersed.
10% of Ru and 100% of I volatilizes and passes out the stack.

Element	Fraction Dispersed	x	Fraction Passing Filter	+	Volatile Fraction	=	RF
Ru	0.01		0.01		0.10		0.10
Zr-Nb	0.01		.01		-		1×10^{-4}
I	-		-		1.0		1.0
OFPT	0.01		0.01		-		1×10^{-4}
Pu	0.01		0.01		-		1×10^{-4}

Equivalence Factor:

Area of the HILC = 1080 ft^2.
Total burned = 100 gallons = 378 liters.
HAF = 2260 liters/MTU during Np campaign.
HAF/HAX = 0.51 during Np campaign.
HAX = HAP = 4431 liters/MTU.

$$EF = \frac{378 \text{ liters}}{4431 \text{ liters/MTU}} = 0.085 \text{ MTU}$$

This incident takes place in the High Intermediate Level Cell (HILC). There are 2 HEPA filters before stack release. The volatile releases are given in Table V-8 and the non-volatile releases are given in Table V-9.

TABLE V-8

VOLATILE RELEASES

Group	Isotope	Ci/Tonne	X	DF	X	SF	X	EF	Volatile Fraction	Stack Release Ci
Ru:						0.01				
	Ru-103	1.2×10^5		0.16				0.085	0.1	1.63EO*
	Ru-106	6.1×10^5		0.73						3.78E1
I:						0.04		0.085	1.0	
	I-129	3.6×10^{-2}		1.0						1.22E-4
	I-131	1.6		0.032						1.74E-4

* 1.63EO = 1.63×10^0

TABLE V-9

NON-VOLATILE RELEASES

										Stack Release Ci		
Group	Isotope	Ci/Tonne	X	DF**	X	SF	X	EF	Fraction Dispersed	No Filter Works	One Filter Works	Two Filters Works
Ru:						0.01		0.085				
	Ru-103	1.2x10^{5}		0.16					.01	1.63E-1	1.63E-4	1.63E-6
	Ru-106	6.1x10^{5}		0.73						1.63E-1	1.63E-4	1.63E-6
										3.78EO	3.78E-3	3.78E-5
Zr-Nb:						0.01		0.085				
	Zr-95	3.5x10^{5}		0.14					.01	4.17E-1	4.17E-4	4.17E-6
	Nb-95	6.5x10^{5}		0.14						7.74E-1	7.74E-4	7.74E-6
I:						0.04		0.085				
	I-129	3.6x10^{-2}		1.0					0	0	0	0
	I-131	1.6		0.032						0	0	0
OFPT;						0.001		0.085				
	Sr-89	9.0x10^{4}		0.20					.01	1.53E-2	1.53E-5	1.53E-7
	Sr-90	8.4x10^{4}		1.0						7.13E-2	7.13E-5	7.13E-7
	Y-90	8.4x10^{4}		1.0						7.13E-2	7.13E-5	7.13E-7
	Y-91	1.9x10^{5}		0.23						3.71E-2	3.71E-5	3.71E-7
	Cs-134	2.4x10^{5}		0.79						1.61E-1	1.61E-4	1.61E-6
	Cs-137	1.2x10^{5}		0.96						9.78E-2	9.78E-5	9.78E-7
	Ce-141	7.9x10^{4}		0.13						8.74E-3	8.74E-6	8.74E-8
	Ce-144	8.8x10^{5}		0.66						4.94E-1	4.94E-4	4.94E-6
	Pm-147	1.4x10^{5}		0.90						1.07E-1	1.07E-4	1.07E-6
	Am-241	250.		1.0						2.12E-4	2.12E-7	2.12E-9
	Am-242	4.0		1.0						3.4 E-6	3.4 E-9	3.4 E-11
	Cm-242	4.4x10^{4}		0.43						1.6 E-2	1.6 E-5	1.6 E-7
	Cm-243	34.0		0.99						2.86E-5	2.86E-8	2.86E-10
	Cm-244	5.7x10^{3}		1.0						4.84E-3	4.84E-6	4.84E-8
Pu:						0.01		0.085				
	Pu-238	4.3x10^{3}		1.0					.01	3.65E-2	3.65E-5	3.65E-7
	Pu-239	3.2x10^{2}		1.0						2.72E-3	2.72E-6	2.72E-8
	Pu-240	6.3x10^{2}		1.0						5.35E-3	5.35E-6	5.35E-8
	Pu-241	1.7x10^{5}		1.0						1.44EO	1.44E-3	1.44E-5
	Pu-242	3.6		1.0						3.06E-5	3.06E-8	3.06E-10

4. Explosion in the LAW Concentrator

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The Barnwell[7] calculation is given in Table V-10.

TABLE V-10 - SAR Results

Group*	Isotope:	Ci/Tonne	× SF	× RF	÷ EF	= Ci Released
Ru:			0.02	0.001	3.5	
	Pu-103	1.2×10^{5}				8.6
	Ru-106	6.1×10^{5}				43.0
Zr-Nb:			0.02	1.1×10^{-7}	3.5	
	Zr-95	3.5×10^{5}				.0028
	Nb-95	6.5×10^{5}				.0052
I:			0.032	1.0	3.5	
	I-129	3.6×10^{-2}				4.0×10^{-3}
	I-131	1.6				0.18
OFPT:			0.002	1.1×10^{-7}	3.5	
	Sr-89	9.0×10^{4}				7.0×10^{-5}
	Sr-90	8.4×10^{4}				6.6×10^{-5}
	Y-90	8.4×10^{4}				6.6×10^{-5}
	Y-91	1.9×10^{5}				1.5×10^{-4}
	Cs-134	2.4×10^{5}				1.8×10^{-4}
	Cs-137	1.2×10^{5}				9.4×10^{-5}
	Ce-141	7.9×10^{4}				6.2×10^{-5}
	Ce-144	8.8×10^{5}				6.9×10^{-4}
	Pm-147	1.4×10^{5}				1.1×10^{-4}
	Am-241	250.				Nil
	Am-242	4.0				Nil
	Cm-242	4.4×10^{4}				3.5×10^{-5}
	Cm-243	34.0				2.7×10^{-7}
	Cm-244	5.7×10^{3}				4.5×10^{-6}
Pu:			9×10^{-4}	1.1×10^{-7}	3.5	
	Pu-238	4.3×10^{3}				1.5×10^{-6}
	Pu-239	3.2×10^{2}				1.1×10^{-7}
	Pu-240	6.3×10^{2}				2.2×10^{-7}
	Pu-241	1.7×10^{5}				6.0×10^{-5}
	Pu-242	3.6				Nil

*OFPT = Other fission products and transuranium elements.

Release factor:

Non-volatile fraction dispersed:

Concentrator volume = 1500 liters of concentrate
Cell volume = 1510 m^3
Aerosol content = 100 mg/m^3
Concentrate dispersed = 1.5×10^{5} mg = 0.15 kg
Concentrate density = 1.3 kg/ℓ
Non-volatile fraction dispersed = 8×10^{-5}
Dispersion passing through filter = 0.14 mg/m^3
Filter factor = 1.4×10^{-3}

Element	Non-Vol. Fraction Dispersed ×	Filter Factor +	Volatile Fraction =	RF
Ru	8×10^{-5}	1.4×10^{-3}	.001	0.001
Zr-Nb	8×10^{-5}	1.4×10^{-3}	0	1.1×10^{-7}
I			1.0	1.0
OFPT	8×10^{-5}	1.4×10^{-3}	0	1.1×10^{-7}
Pu	8×10^{-5}	1.4×10^{-3}	0	1.1×10^{-7}

Equivalence factor:

Concentrate [illegible] 90 ℓ/m [illegible] ℓ/M[illegible] ([illegible] Pu fuel)
EF - 3.4 MTU

The LAW concentrator is in the High Intermediate Level Cell (HILC) and there are 2 HEPA filters prior to stack release. Table V-11 lists the volatile releases and Table V-12 lists the non-volatile releases.

TABLE V-11

VOLATILE RELEASES

Group	Isotope	Ci/Tonne	X	SF	X	EF	Volatile Fraction	Stack Release Ci
Ru:				0.02		3.5	.001	
	Ru-103	1.2×10^{5}						8.4EO
	Ru-106	6.1×10^{5}						4.27E1
I:				0.032		3.5	1.0	
	I-129	3.6×10^{-2}						4.03E-3
	I-131	1.6						1.79E-1

*8.4EO = 8.4 x 10^{0}

TABLE V-12

NON-VOLATILE RELEASES

Group*	Isotope:	Ci/Tonne	X	SF	X	EF	Fraction Dispersed	Stack Release Ci No Filter Works	One Filter Works	Two Filters Works
Ru:				0.02		3.5	8.0E-5*			
	Ru-103	$1.2x10^{5}$						6.72E-1	6.72E-4	6.72E-6
	Ru-106	$6.1x10^{5}$						3.42EO	3.42E-3	3.42E-5
Zr-Nb:				0.02		3.5	8.0E-5			
	Zr-95	$3.5x10^{5}$						1.96EO	1.96E-3	1.96E-5
	Nb-95	$6.5x10^{5}$						3.64EO	3.64E-3	3.64E-5
I:				0.032		3.5	0			
	I-129	$3.6x10^{-2}$						0	0	0
	I-131	1.6						0	0	0
OFPT:				0.002		3.5	8.0E-5			
	Sr-89	$9.0x10^{4}$						5.02E-2	5.02E-5	5.02E-7
	Sr-90	$8.4x10^{4}$						4.7 E-2	4.7 E-5	4.7 E-7
	Y-90	$8.4x10^{4}$						4.7 E-2	4.7 E-5	4.7 E-7
	Y-91	$1.9x10^{5}$						1.06E-1	1.06E-4	1.06E-6
	Cs-134	$2.4x10^{5}$						1.34E-1	1.34E-4	1.34E-6
	Cs-137	$1.2x10^{5}$						6.72E-2	6.72E-5	6.72E-7
	Ce-141	$7.9x10^{4}$						4.42E-2	4.42E-5	4.42E-7
	Ce-144	$8.8x10^{5}$						4.83E-1	4.83E-4	4.83E-6
	Pm-147	$1.4x10^{5}$						7.84E-2	7.84E-5	7.84E-7
	Am-241	250.						1.40E-4	1.40E-7	1.40E-9
	Am-242	4.0						2.24E-6	2.24E-9	2.24E-11
	Cm-242	$4.4x10^{4}$						2.46E-2	2.46E-5	2.46E-7
	Cm-243	34.0						1.90E-5	1.90E-8	1.90E-10
	Cm-244	$5.7x10^{3}$						3.91E-3	3.19E-6	3.19E-8
Pu:				$9x10^{-4}$		3.5	8.0E-5			
	Pu-238	$4.3x10^{3}$						1.08E-3	1.08E-6	1.08E-8
	Pu-239	$3.2x10^{2}$						8.06E-5	8.06E-8	8.06E-10
	Pu-240	$6.3x10^{2}$						1.59E-4	8.06E-8	8.06E-10
	Pu-241	$1.7x10^{5}$						4.28E-2	4.28E-5	4.28E-7
	Pu-242	3.6						9.07E-7	9.07E-10	9.07E-12

* 8.0E-5 = 8.0×10^{-5}

5. Ion Exchange Resin Fire (Neptunium Recovery Campaign)

The Barnwell[7] Calculation is given in Table V-13.

TABLE V-13

SAR Results

Isotope	Source Activity, Ci/tonne	×	SF	×	RF	×	EF	=	Activity Released, Ci
Ru-103	1.24×10^{5}		3.0×10^{-6}		5.1×10^{-2}		5		9.49×10^{-2}
Ru-106	7.22×10^{5}		3.0×10^{-6}		5.1×10^{-2}		5		0.552
Zr-95	3.58×10^{5}		6.6×10^{-6}		1.0×10^{-3}		5		1.18×10^{-2}
Nb-95	6.82×10^{5}		6.6×10^{-6}		1.0×10^{-3}		5		2.25×10^{-2}
I-129	3.55×10^{-2}		3.1×10^{-7}		.50		5		2.75×10^{-8}
I-131	1.6		3.1×10^{-7}		.50		5		1.24×10^{-6}
Sr-89	8.2×10^{4}		1.0×10^{-8}		1.0×10^{-3}		5		4.10×10^{-6}
Sr-90	8.2×10^{4}		1.0×10^{-8}		1.0×10^{-3}		5		4.10×10^{-6}
Cs-134	1.79×10^{5}		1.0×10^{-8}		1.0×10^{-3}		5		8.95×10^{-6}
Cs-137	1.25×10^{5}		1.0×10^{-8}		1.0×10^{3}		5		6.25×10^{-6}
Ba-137m	1.10×10^{5}		1.0×10^{-8}		1.0×10^{-3}		5		5.50×10^{-6}
Ce-144	5.6×10^{5}		1.0×10^{-8}		1.0×10^{-3}		5		2.80×10^{-5}
Cm-242	4.0×10^{4}		1.0×10^{-8}		1.0×10^{-3}		5		2.00×10^{-6}
Cm-244	4.9×10^{3}		1.0×10^{-8}		1.0×10^{-3}		5		2.45×10^{-7}
Np-238	6.38		0.95		1.0×10^{-3}		5		3.03×10^{-2}
Pu-238	4.37×10^{3}		5.0×10^{-4}		1.0×10^{-3}		5		1.09×10^{-2}
Pu-239	3.2×10^{2}		5.0×10^{-4}		1.0×10^{-3}		5		8.00×10^{-4}
Pu-240	6.3×10^{2}		5.0×10^{-4}		1.0×10^{-3}		5		1.58×10^{-3}
Pu-241	1.6×10^{5}		5.0×10^{-4}		1.0×10^{-3}		5		0.40

Element	Non-Volatile Fraction Passing Filter	+	Volatile Fraction	=	RF
Ru	9.0×10^{-4}		.05		5.1×10^{-2}
Zr-Nb	1.0×10^{-3}		0		1.0×10^{-3}
I	0		.50		.50
Sr, Ce, Cs, Ba, Cm	1.0×10^{-3}		0		1.0×10^{-3}
Pu	1.0×10^{-3}		0		1.0×10^{-3}
Np	1.0×10^{-3}		0		1.0×10^{-3}

Equivalence Factor (EF)

EF=(5MTU/day)(1 day) = 5MTU

The accident occurs in the Plutonium Product Cell (PPC) where there are three HEPA's in series before stack release. 10% of the Ru and 100% of the I are volatilized during the burning. The results are given below for the volatile and non-volatile releases. These results are included for completeness. They are not included in risk assessment considerations since it is anticipated that this unit operation will not be used in the reprocessing plants under consideration.

TABLE V-14

VOLATILE SOURCE

Isotope	Source Activity Ci/tonne	X	SF	X	EF	Volatile Fraction	Stack Release Ci
Ru-103	1.24×10^5		3.0×10^{-6}		5	.05	**9.3E-2***
Ru-106	7.22×10^5		3.0×10^{-6}		5	.05	**5.42E-1**
I-129	3.55×10^{-2}		3.1×10^{-7}		5	.5+	**2.75E-8**
I-131	1.6		3.1×10^{-7}		5	.5	**1.24E-6**

*** 9.3E-2 = 9.3×10^{-2}**

+ Reference 5

TABLE V-15 NON-VOLATILE SOURCE

					Stack Release Ci			
Isotope	Source Activity Ci/tonne	SF	EF	Fraction Dispersed	No Filter Works	One Filter Works	Two Filters Work	Three Filte Work
Ru-103	1.24×10^{5}	3.0×10^{-6}	5	.45	8.37E-1*	8.37E-4	8.37E-6	5.02E-7
Ru-106	7.22×10^{5}	3.0×10^{-6}	5	.45	4.87EO	4.87E-3	4.87E-5	2.92E-6
Zr-95	3.58×10^{5}	6.6×10^{-6}	5	.5	5.91EO	5.91E-3	5.91E-5	3.54E-6
Nb-95	6.82×10^{5}	6.6×10^{-6}	5	.5	1.13EO	1.13E-2	1.13E-4	6.78E-6
I-129	3.55×10^{-2}	3.1×10^{-7}	5	0	0	0	0	0
I-131	1.6	3.1×10^{-7}	5	0	0	0	0	0
Sr-89	8.2×10^{4}	1.0×10^{-8}	5	.5	2.05E-3	2.05E-6	2.05E-8	1.23E-9
Sr-90	8.2×10^{4}	1.0×10^{-8}	5	.5	2.05E-3	2.05E-6	2.05E-8	1.23E-9
Cs-134	1.79×10^{5}	1.0×10^{-8}	5	.5	4.48E-3	4.48E-6	4.48E-8	2.69E-9
Cs-137	1.25×10^{5}	1.0×10^{-8}	5	.5	3.13E-3	3.13E-6	3.13E-8	1.88E-9
Ba-137m	1.10×10^{5}	1.0×10^{-8}	5	.5	2.75E-3	2.75E-6	2.75E-8	1.65E-9
Ce-144	5.6×10^{5}	1.0×10^{-8}	5	.5	1.40E-2	1.40E-5	1.40E-7	8.40E-9
Cm-242	4.0×10^{4}	1.0×10^{-8}	5	.5	1.0 E-3	1.0 E-6	1.0 E-8	6.0 E-10
Cm-244	4.9×10^{3}	1.0×10^{-8}	5	.5	1.18E-4	1.18E-7	1.18E-9	7.08E-11
Np-238	6.38	0.95	5	.5	1.52E1	1.52E-2	1.52E-4	9.12E-6
Pu-238	4.37×10^{3}	5.0×10^{-4}	5	.5	5.47EO	5.47E-3	5.47E-5	3.28E-6
Pu-239	3.2×10^{2}	5.0×10^{-4}	5	.5	4.0E-1	4.0E-4	4.0E-6	2.40E-7
Pu-240	6.3×10^{2}	5.0×10^{-4}	5	.5	7.88E-1	7.88E-4	7.88E-6	4.37E-7
Pu-241	1.6×10^{5}	5.0×10^{-4}	5	.5	2.0E2	2.0E-1	2.0E-3	1.20E-4

* 8.37E-1 = 8.37×10^{-1}

6. Nuclear Criticality Incident

The Barnwell[7] results for a criticality incident in the Remote Process Cell (RPC) are given in Table V-16.

TABLE V-16

SAR Results

NUCLEAR CRITICALITY INCIDENT (10^{18} Fissions)

Isotope	Activity Released,* Ci@t=0
I-131(8.05d)	0.75
I-132(2.4h)	3.30
I-133(20.5h)	18.0
I-134(52.5m)	450.
I-135(6.68hr)	48.0
Xe-135m(15m)	395.
Xe-138(17m)	1050.
Kr-87(1.3h)	112.
Kr-83m(1.86h)	13.5
Kr-88(2.8h)	69.5
Kr-85m(4.4h)	18.5
Xe-135(9.2h)	36.4
Xe-133m(2.3d)	0.20
Xe-133(5.27d)	2.70
Xe-131m(12.0d)	6.0×10^{-2}
Kr-85(10.4y)	2.0×10^{-3}

*The iodine is assumed to be volatile. Of the amount generated, 1% is assumed to be released to the vent system, the rest remaining in solution.

7. Explosion in the HAF Tank

This accident is assumed to occur in the High Level Cell (HLC). The consequences of this accident have been evaluated assuming that there is only one HEPA filter before stack release. The release data are given in Tables V-17 and V-18.

> Note: It is more likely that for present and future plant designs that at least two filters will be in line between the HLC and the stack.

TABLE V-17

VOLATILE SOURCE

Group	Isotope	Ci/Tonne	Split Factor	Equivalence Factor	Volatile Fraction	Stack Release Ci
Ru:			1.0	4	.001	
	Ru-103	1.2×10^{5}				4.8E2*
	Ru-106	6.1×10^{5}				2.44E3
I:			0.1	4	1.0	
	I-129	3.6×10^{-2}				1.4E-2
	I-131	1.6				6.4E-1

*4.8E2 = 4.8×10^{2}

TABLE V-18 NON-VOLATILE SOURCE

Group	Isotope	Ci/Tonne	Split Factor	Equivalence Factor	Fraction Dispersed	Stack Release Ci No Filter Works	One Filter Works
Ru:			1.0	4	5.9E-6*		
	Ru-103	$1.2\text{x}10^5$				2.83EO	2.83E-3
	Ru-106	$6.1\text{x}10^5$				1.44E1	1.44E-2
ZR-Nb:			1.0	4	5.9E-6		
	Zr-95	$3.5\text{x}10^5$				8.26EO	8.26E-3
	Nb-95	$6.5\text{x}10^5$				1.53E1	1.53E-2
I:			0.5	4	0		
	I-129	$3.6\text{x}10^{-2}$				0	
	I-131	1.6				0	
CFPZ:			1.0	4	5.9E-6		
	Sr-89	$9.0\text{x}10^4$				2.12EO	2.12E-3
	Sr-90	$8.4\text{x}10^4$				1.98EO	1.98E-3
	Y-90	$8.4\text{x}10^4$				1.98EO	1.98E-3
	Y-91	$1.9\text{x}10^5$				4.48EO	4.48E-3
	Cs-134	$2.4\text{x}10^5$				5.66EO	5.66E-3
	Cs-137	$1.2\text{x}10^5$				2.83EO	2.83E-3
	Ce-141	$7.9\text{x}10^4$				1.86EO	1.86E-3
	Ce-144	$8.8\text{x}10^5$				2.08E1	2.08E-2
	Pm-147	$1.4\text{x}10^5$				3.31EO	3.31E-3
	Am-241	250.				5.90E-3	5.90E-6
	Am-242	4.0				9.45E-5	9.45E-8
	Cm-242	$4.4\text{x}10^4$				1.04EO	1.04E-3
	Cm-243	34.0				8.03E-4	8.04E-7
	Cm-244	$5.7\text{x}10^3$				1.35E-1	1.35E-4
Pu:			1.0	4	5.9E-6		
	Pu-238	$4.3\text{x}10^3$				1.02E-1	1.02E-4
	Pu-239	$3.2\text{x}10^2$				7.55E-3	7.55E-6
	Pu-240	$6.3\text{x}10^2$				1.49E-2	1.49E-5
	Pu-241	$1.7\text{x}10^5$				4.01EO	4.01E-3
	Pu-242	3.6				8.50E-5	8.50E-8

* 5.9E-6 = 5.9×10^{-6}

8. Waste Calciner Explosion

The calciner would be fed from the High Aqueous Waste (HAW) concentrator. It would probably be located in the same cell area, which in the case of Barnwell, is the Remote Process Cell (RPC). The calciner operates at several hundred degrees Celsius and this would raise the amount of Ru volatilized during an accident by about a factor of 10 over the HAW concentrator explosion results. Aside from this variation, the results of a waste calciner explosion would be essentially what they were in the HAW concentrator explosion analysis.

9. Fuel Receiving and Storage Accident

Incidents that release radioactivity in the receiving and storage area are varied. A scenario which may bound many similar incidents in the nature and magnitude of the release was chosen for evaluation.

While being shipped the spent fuel cask has lost its heat removal capability. The fuel rods fail the clad and release a large fraction of the more volatile fission products. When the cask is opened in the receiving and storage area it will be submerged in water. The cask interior may be dry and thus, much of the fission activity leaves the cask in a large stream of bubbles that rise to the surface. Some of the fission products will have plated out on the cask interior walls and some will remain in the pool water. The released products will either enter the Fuel Receiving and Storage area (FRS) atmosphere or they may enter the DOG/VOG system.

Table V-19 lists some expected release data for Ru and I. The tritium and noble gas releases are neglected as being part of the normally accepted release. The material quantities used were taken from the data for rail shipments given in Table V-20.

TABLE V-19

Expected Release Data[50,51]

Group	Isotope:	Ci/Tonne	Source/4.5 tonne Shipment train	Building Fraction Released	Ci Released
Ru:					
	Ru-103	1.2×10^{5}	5.4×10^{5}	10%	5.4E4
	Ru-106	6.1×10^{5}	27.4×10^{5}		2.74E5
I:					
	I-129	3.6×10^{-2}	16.2×10^{-2}	10%	1.62E-2
	I-131	1.6	7.2		7.2 E-1
OFP*	Cs-134	2.4×10^{5}	10.8×10^{5}	1%	1.08E4
	Cs-137	1.2×10^{5}	5.4×10^{5}		5.4E3

*OFP = Other Fission Products

TABLE V-20

Volatile Isotope Activity in Spent Fuel Shipments[50]

Isotope	Type of Shipment	MTU per Shipment	Total Activity per Shipment	% Released To Plenum	Activity In Plenum
Kr-85	Truck	0.45	4350	17	740
H-3	Truck		187	1	1.87
I-131	Truck		0.64	2.3	0.015
Kr-85	Rail	4.5	43,500	17	7400
H-3	Rail		1,870	1	18.7
I-131	Rail		6.4	2.3	0.15
Kr-85	Water	18	174,000	17	29,600
H-3	Water		7,480	1	75
I-131	Water		25.6	2.3	0.59

The probability of cask heat transfer failure during shipment was estimated at 10^{-3} to 10^{-4} per shipment. With approximately 250 shipments per year, this became 2.5×10^{-1} to 2.5×10^{-2} per year likelihood. We chose 10^{-1} per year for a 4.5 tonne delivery of failed fuel.

Prior to the cask being opened it was assumed that 100% of the I, 10% of the Ru and 1% of the Cs was volatile. The amount that rises through the pool after the cask is opened under water was assumed as 10% of what was initially volatile for the Ru and I and 0.1% for the Cs.

The released gases subsequently pass through an iodine scrubber (VOG) and then through 2 HEPA filters. It was assumed that 7% of the Ru, 0.01% of the I and 0.1% of the Cs would pass through the iodine scrubber. It was also assumed that the passage through the iodine scrubber would cause the volatiles to become non-volatile. Hence the HEPA filter failure probabilities are taken as 10^{-2} and 10^{-3} per demand, as previously noted. Table V-21 presents the results. Noble gases were considered as normal releases.

TABLE V-21

FUEL RECEIVING AND STORAGE ACCIDENT

Isotope	Ci in a 4.5 tonne Shipment	Amt. in Cask Atmosphere Before Release	Amt. in FRSS Atmos.after Cask is Opened	Amt.After Iodine Scrubber	Amt.After 1 Filter	Amt. After 2 Filter
Ru 103	$5.4\text{x}10^5$ (.10)*	5.4E4 (.1)*	5.4E3 (.07)*	3.78E2	3.78E-1	3.78E-3
Ru 106	$2.74\text{x}10^6$ (.10)	2.74E5 (.1)	2.74E4 (.07)	1.92E3	1.92E0	1.92E-2
I 129	$1.62\text{x}10^{-1}$ (1.0)	1.62E-1 (.1)	1.62E-2 (10^{-4})	1.62E-6	1.62E-6	1.62E-6
I 131	7.2 (1.0)	7.2E0 (.1)	7.2E-1 (10^{-4})	7.2E-5	7.2E-5	7.2E-5
Cs 134	$1.08\text{x}10^6$ (.01)	1.08E4 (.001)	1.08E1 (10^{-3})	1.08E-2	1.08E-5	1.08E-7
Cs 137	$5.4\text{x}10^5$ (.01)	5.4E3 (.001)	5.4E0 (10^{-3})	5.4E-3	5.4E-6	5.4E-8

* Fraction available for release

VI. Risk Assessment

1. Release Likelihood Spectra

To quantify the risk from a generic reprocessing plant requires the synthesis of accident likelihoods and their consequences. In Section 3, a number of fault trees keyed to each process cell were constructed. While these fault trees are generic in nature they do indicate the complications that are required in order to have an accident and also indicate the probable likelihood of such an event. In Section 5, the consequences from each of these accident sequences identified in the fault trees are evaluated. In this section, the results of Sections 3 and 5 are combined to produce a spectrum of release likelihood curves for a variety of isotopes.

Since there are 10 to 15 accidents that were considered, an alphabetical code is employed in plotting up the data for accident identification. This alphabetical code is given in Table VI-1. Also shown in Table VI-1 are the number of HEPA filters normally found between the process cell of interest and the final exhaust stack. Both the process cell and the fault trees that were utilized to evaluate the likelihood of the incident occurring in the cell are given in the table as well as the probability of occurrence of the incident.

Figures VI-1 through VI-5 are plots of the data for the classes of isotopes considered. Iodine and ruthenium are

TABLE VI-1

ACCIDENTS CONSIDERED IN PRESENT ANALYSIS

ACCIDENT	LOCATION	NO. OF HEPA FILTERS	TREES	LIKELIHOOD/YEAR
A. HAW Conc Explosion	RPC	2	RPC, S, RO	10^{-5}
B. Solvent Fire Pu Extraction Cycle	PPC	3	PPC, SF	10^{-6}
C. Solvent Fire Codecontamination Cycle	HILC	2	HILC, SF	10^{-6}
D. LAW Conc Explosion	ILC	2	ILC, S, RO	10^{-4}
E. Ion Exchange Resin Fire (Not Plotted)	PPC	3	PPC, IER	10^{-4}
F. Criticality Incident	Various (RPC Typical)	– 2	RPC, CP	10^{-5}
G. HAF Tank Explosion	HLC	1*	HLC, H	10^{-5}
H. Waste Calciner Explosion	(RPC)	2	RPC, S, RO, H	10^{-6}
I. Fuel Receiving and Storage Accident	FRSS		FRS	10^{-1}

* The analysis presented in this report assumes that there is only one HEPA filter between HLC and stack. However, it is more likely that for present and future plant designs that at least two filters will be in line between HLC and the stack.

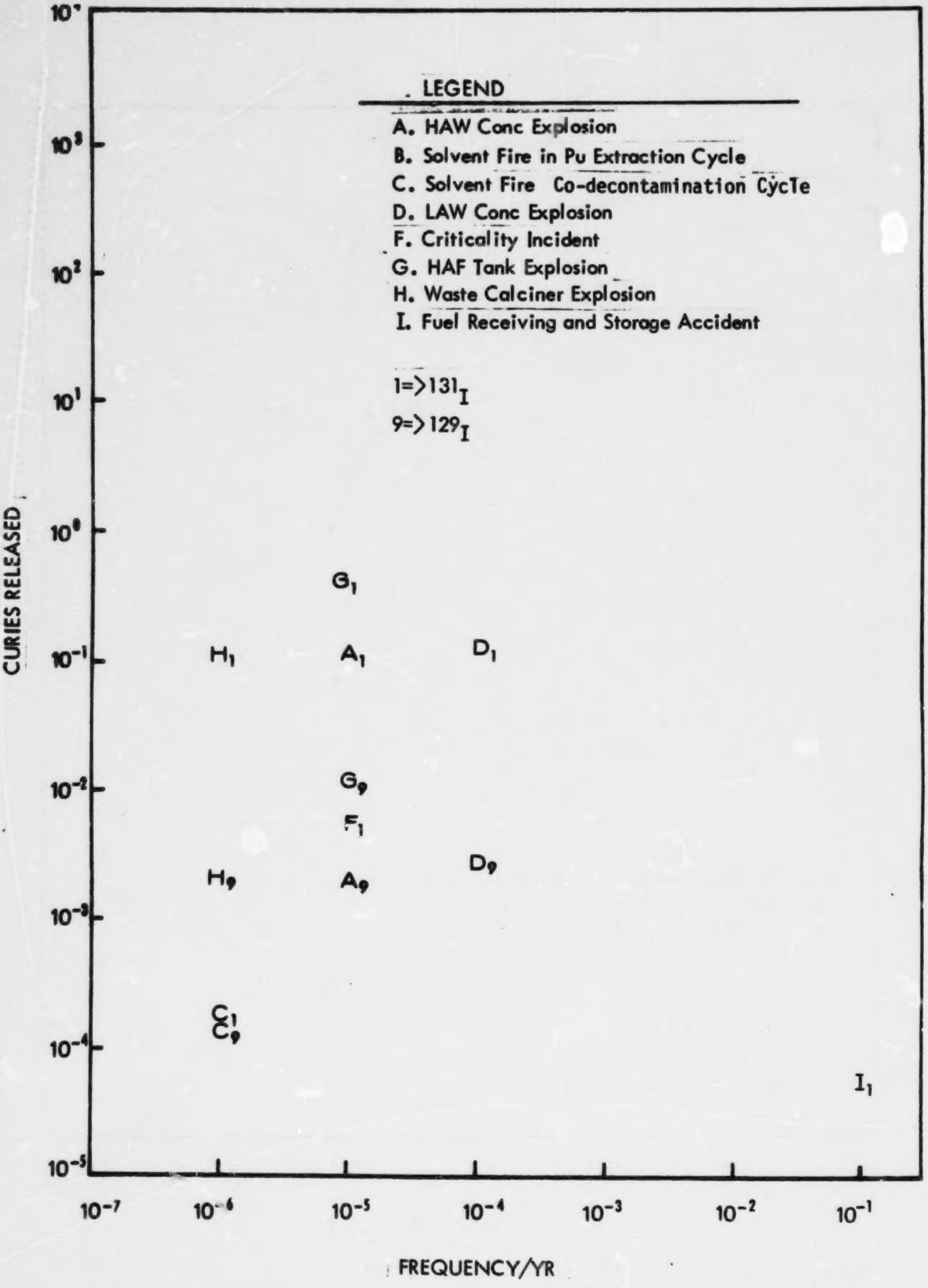

Figure VI -1 IODINE RELEASES ANTCIPATED FOR THE HYPOTHETICAL ACCIDENTS ANALYZED

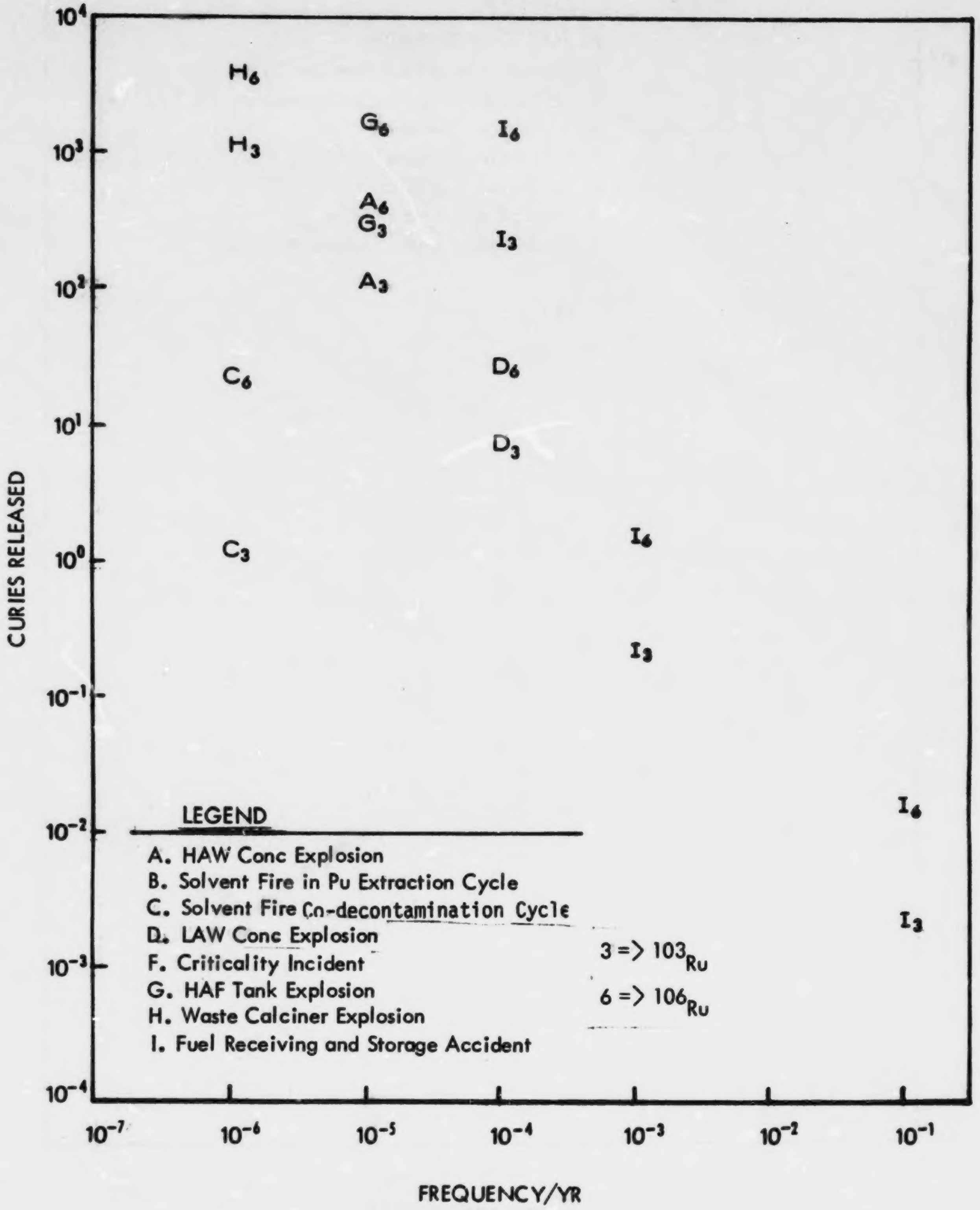

Figure VI-2 RUTHENIUM RELEASES ANTICIPATED FOR THE HYPOTHETICAL ACCIDENTS ANALYZED

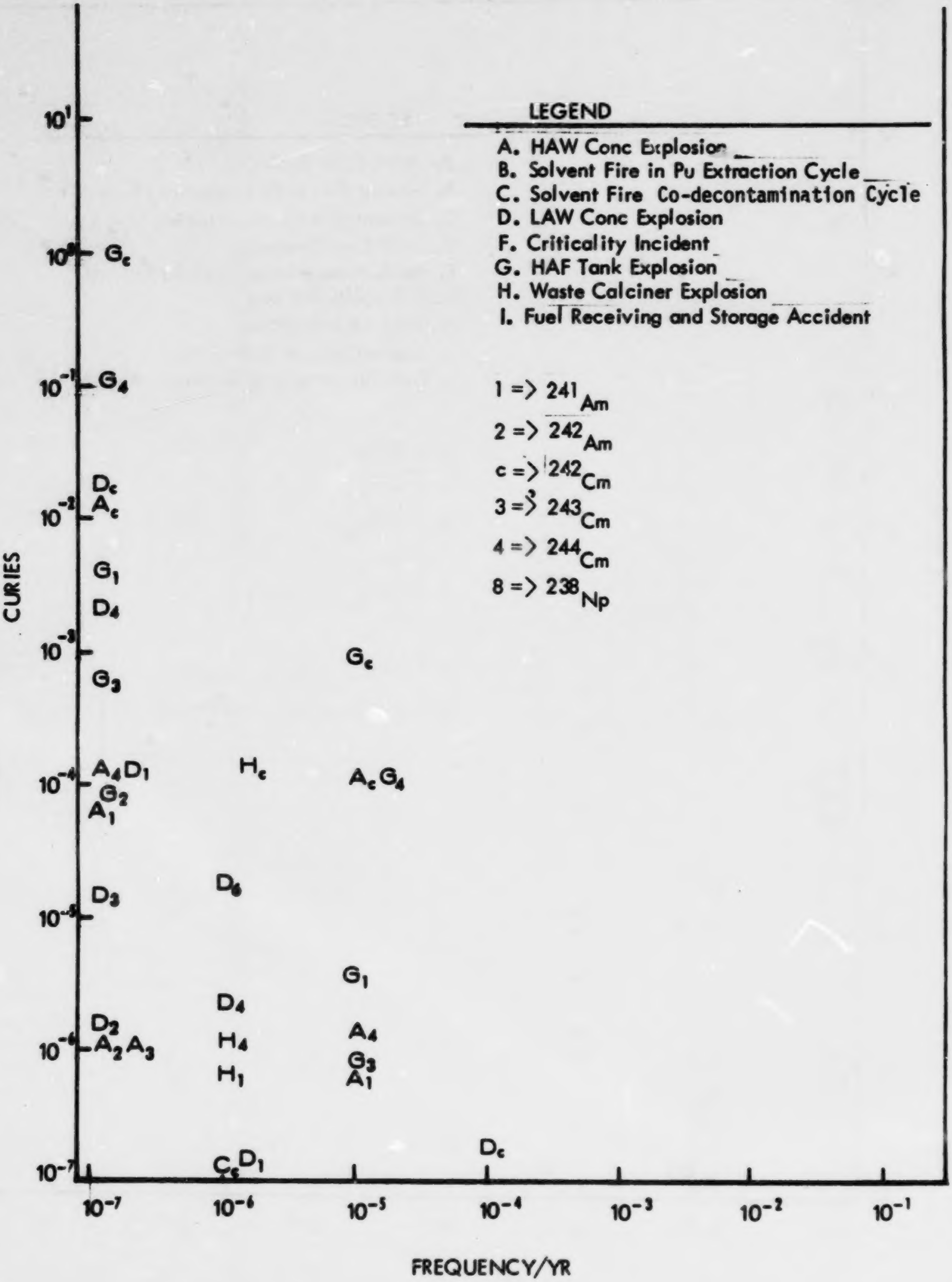

Figure VI-3 NON-VOLATILE ACTINIDE RELEASES ANTICIPATED FOR THE HYPOTHETICAL ACCIDENTS ANALYZED

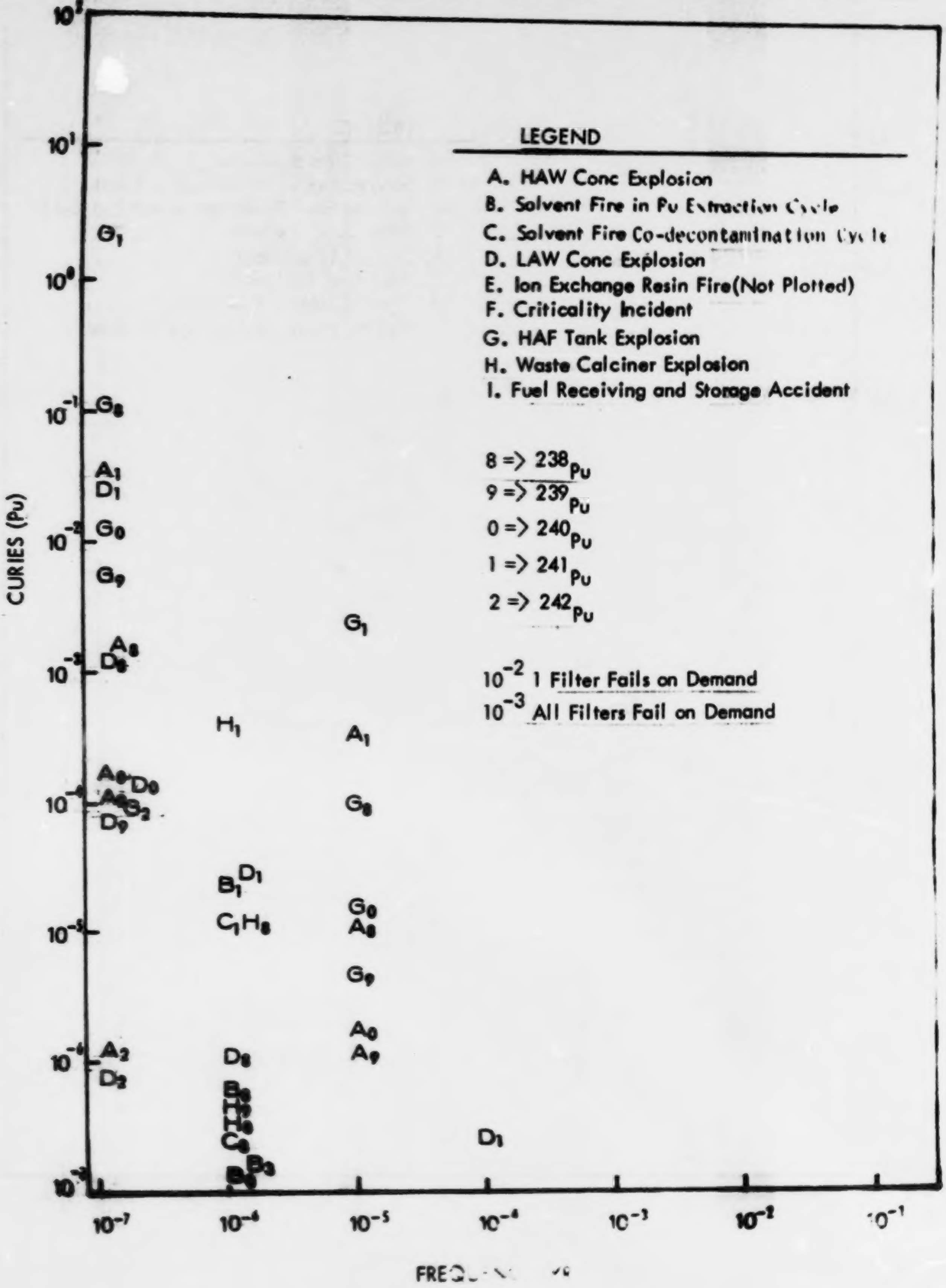

Figure VI-4 PLUTONIMUM RELEASES ANTICIPATED FOR THE HYPOTHETICAL ACCIDENTS ANALYZED

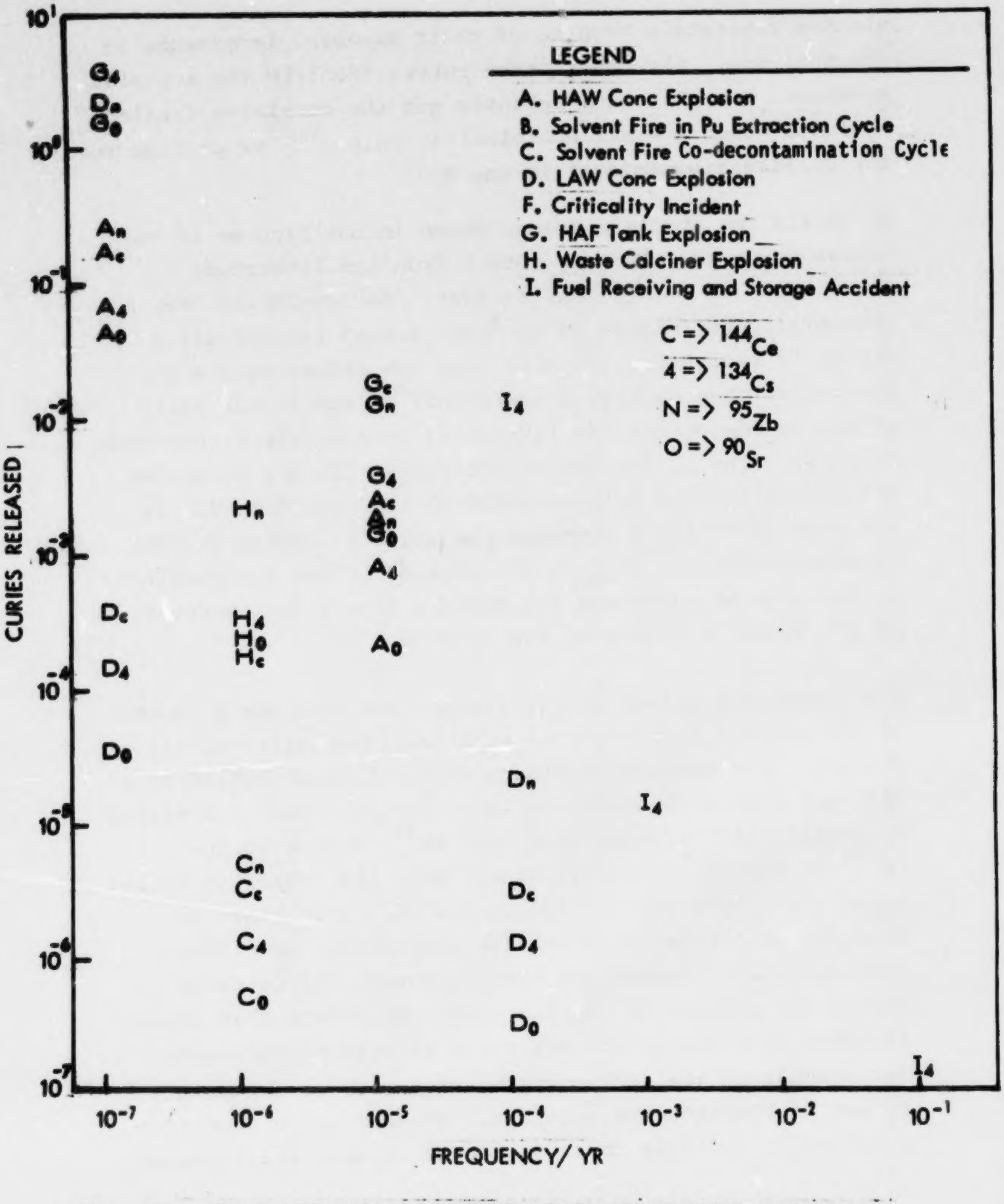

Figure VI-5 OTHER NONVOLATILE FISSION PRODUCT RELEASES ANTICIPATED FOR THE HYPOTHETICAL ACCIDENTS ANALYZED

plotted separately because of their expected importance in the analysis. The other three curves identify the actinide releases, the plutonium releases and the remaining fission products which are all nonvolatile. Only I^{131} is plotted for the criticality incident in the RPC.

To obtain the abscissa values shown in the figures it was necessary to calculate or obtain from the literature a probability of HEPA filter failure. We have found that a probability of failure of 10^{-2} per demand for the first filter is realistic. If more than one filter occurs in the series then we have assumed that an additional factor of ten is needed for the likelihood of complete filter bank failure. That is for two or three HEPA filters in series the probability of failure would be 10^{-3} per demand. In the case of volatile releases the filters are not assumed to work at all. Hence, in the case of iodine and partially in the case of ruthenium the filters do not work because of the volatile nature of the release.

The likelihood values in the figures are then the products of the numbers in Table VI-1 with the probability of filter failure. For example in Figure VI-3, the probability of a HAF tank explosion occurring is 10^{-5}/year. When the filter is working this appears as G_1 for Am^{241} and as Gc for Cm^{242} at the 10^{-5} vertical line. When the filter is failed these two points are shifted to the 10^{-7}/year line, as seen in the figure, in which the probability of filter failure of 10^{-3}/demand has been factored. Since these curves are plotted on log/log paper the points that appear furthest from the origin are those of most significance. For example in the iodine curve, Figure VI-1, the points G_1 and D_1 dominate the spectrum. These figures therefore graphically indicate those accidents of most significance.

2. Dose Quantification

We have selected some of the data given in Figures VI-1 through VI-5 and have calculated the likelihood of receiving a certain dose at a given distance from the generic reprocessing plant site. To do this we utilized the following simplified meteorological model.[55] For a puff release following an accident we have

$$D = Q \bullet K \bullet (X/Q) \tag{6.1}$$

where

Q is the source strength in curies

K is a dose conversion factor in rem meters^3/curie second

D is the received dose in rem

and where

$$(X/Q) = [\pi \bar{u}\ \sigma y\ \sigma z]^{-1} \exp - [(y^2/2\sigma y^2) + (h^2/2\sigma z^2)] \tag{6.2}$$

with

h = release height or stack height in meters

y = cross wind or off centerline distance in meters

$\bar{u}$ = average wind speed in meters/second

σy= cloud horizontal dispersion with distance, in meters

σz= cloud vertical dispersion with distance, in meters.

The following data were used in this analysis

h = 100 meters

$\bar{u}$ = 2 meters/second

y = 0.

Based upon data given in reference 1 for Category D turbulence, Table VI-2 was developed for (X/Q) as a function of distance from the plant site. K values, obtained from reference 56 are listed in Table VI-3.

TABLE VI-2

Category D Meteorology Parameters

Distance From Source [Meters]	σ_y	σ_z	(X/Q) [sec/meter^3] h=0,y=0	(X/Q) [sec/meter^3] h=100 meters,y=0
100	6.	5.	5.32E-3*	0
500	30.	19.	2.79E-4	2.76E-10
1,000	55.	32.	9.0 E-5	6.74E-7
5,000	220.	90.	8.06E-6	4.29E-6
10,000	400.	114.	3.50E-6	2.38E-6
50,000	1500.	310.	3.43E-7	3.25E-7
100,000	2700.	420.	1.41E-7	1.37E-7

* 5.32E-3 = 5.32×10^{-3}

TABLE VI-3

Isotopic Dose Conversion Factors [(1)]

Group	Isotope	K factors $\left[\frac{\text{rem m}^3}{\text{Ci sec}}\right]$
Ru:		
	Ru-103	15.9
	Ru-106	238
Zr-Nb:		
	Zr-95	47.6
	Nb-95	15.9
I:[(2)]		
	I-129	$4.73(10^5)$
	I-131	$8.5\ (10^4)$
I:[(3)]	I-131	$1.6\ (10^3)$
		$3.8\ (10^2)$

Group	Isotope	K factors $\left[\frac{\text{rem m}^3}{\text{Ci sec}}\right]$
OFPT:		
	Sr-89	47.6
	Sr-90	238
	Y-90	
	Y-91	47.6
	Cs-134	119
	Cs-137	95
	Ce-141	9.5
	Ce-144	238
	Pm-147	15.9
	Am-241	$9.52(10^4)$
	Am-242	
	Cm-242	$6.47(10^4)$
	Cm-243	
	Cm-244	$1.26(10^5)$
Pu:		
	Pu-238	$3.81(10^5)$
	Pu-239	$3.81(10^5)$
	Pu-240	$3.81(10^5)$
	Pu-241	$3.81(10^2)$
	Pu-242	$3.81(10^5)$

(1) Conversion factors for dose to lung from inhalation of insoluble particles except for iodine for which the organ of reference is the thyroid.

(2) Ingestion of milk by infant.

(3) Inhalation by adult.

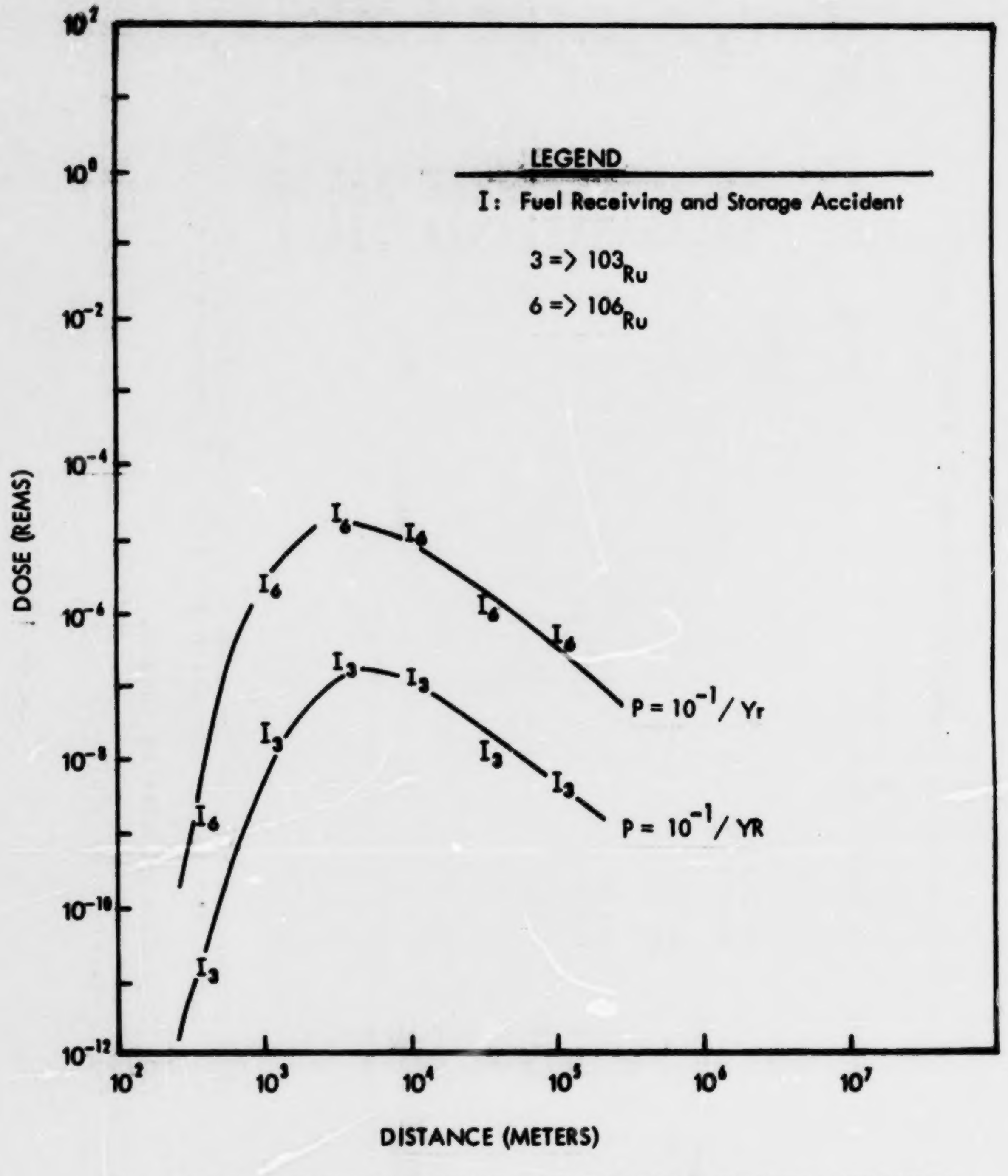

Figure VI-6a RUTHENIUM DOSE AT DISTANCE FOR A CERTAIN LIKELIHOOD

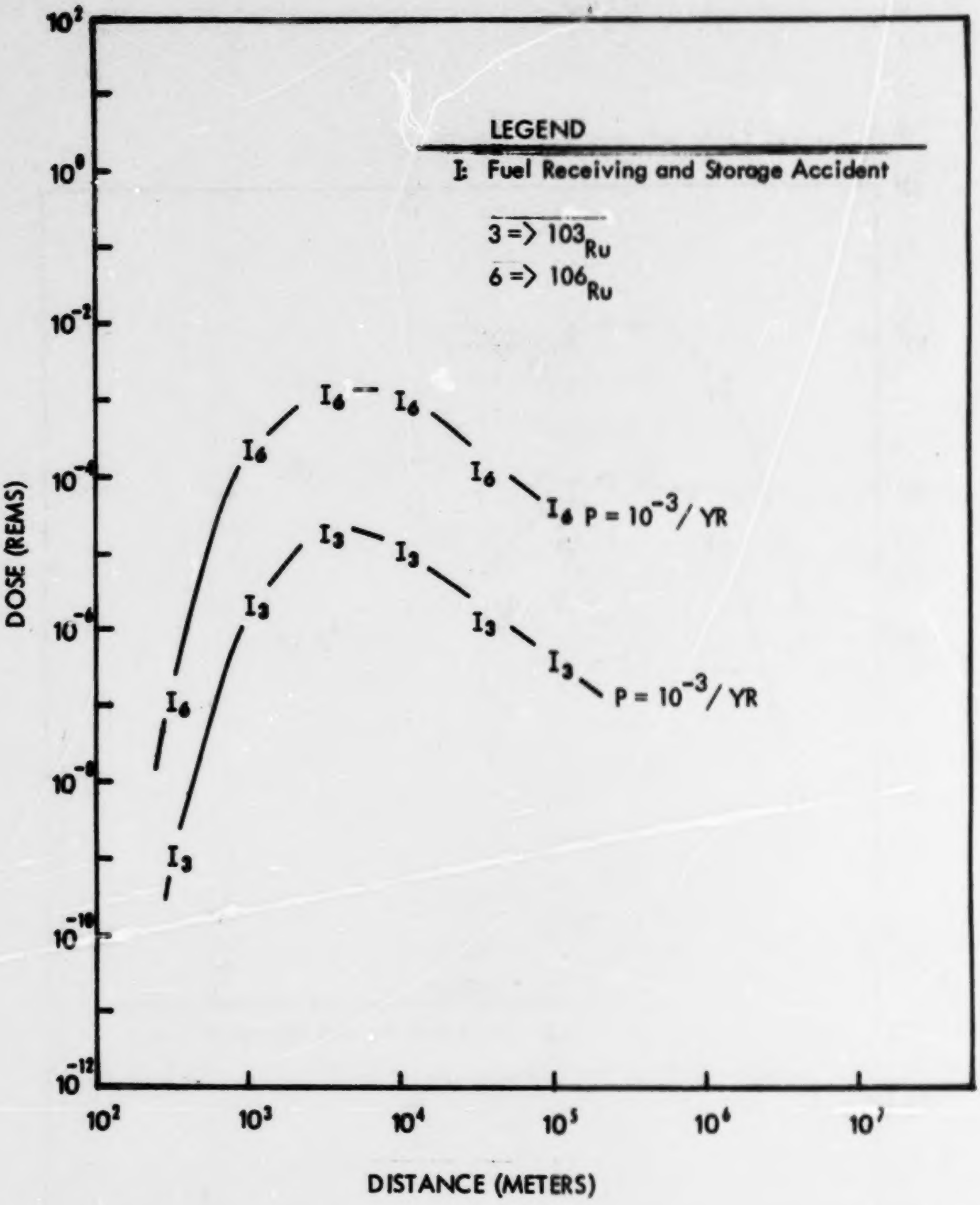

Figure VI-6b RUTHENIUM DOSE AT DISTANCE FOR A LIKELIHOOD OF 10^{-3}/YR

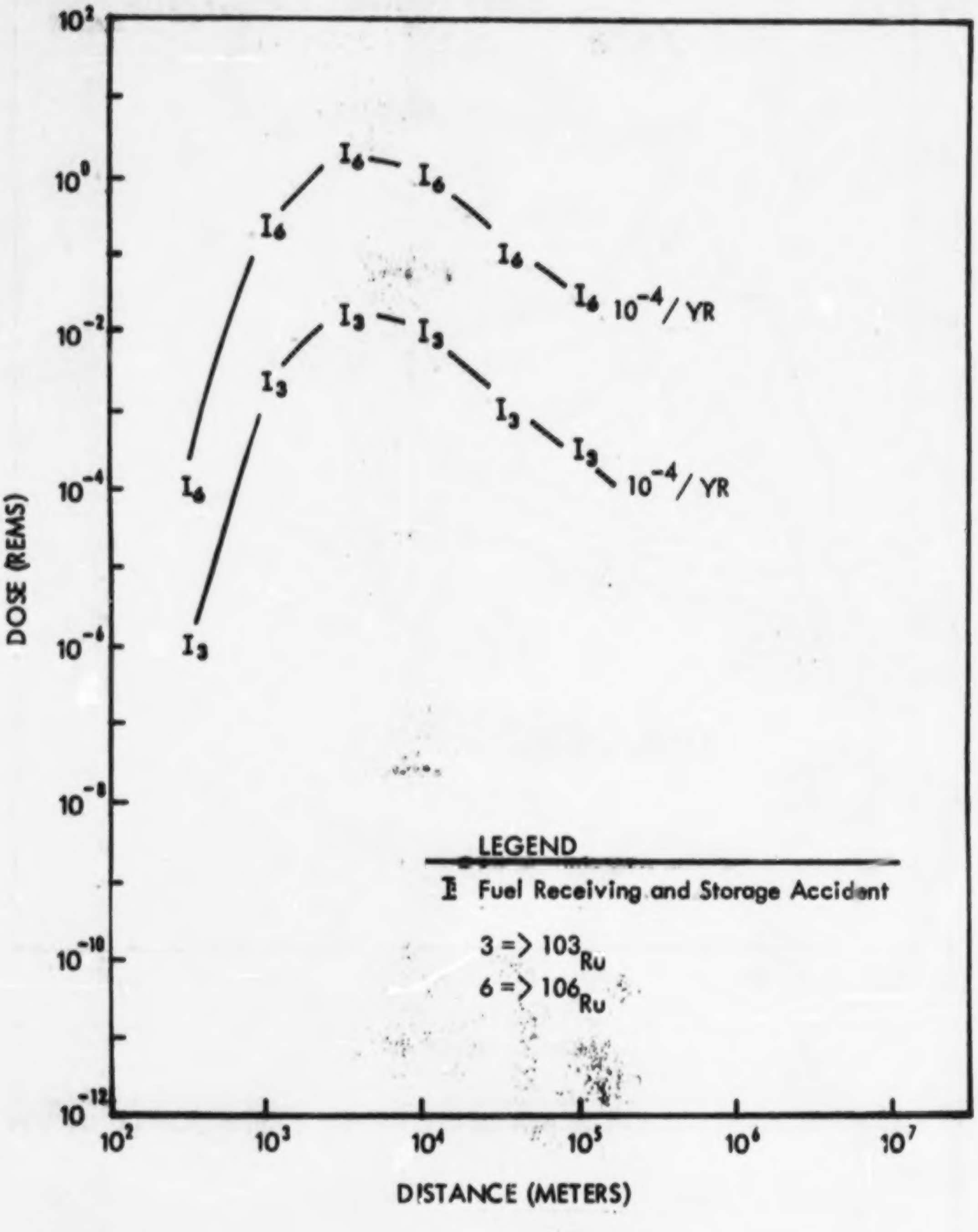

Figure VI-6c RUTHENIUM DOSE AT DISTANCE FOR A LIKELIHOOD OF 10^{-4}/YR

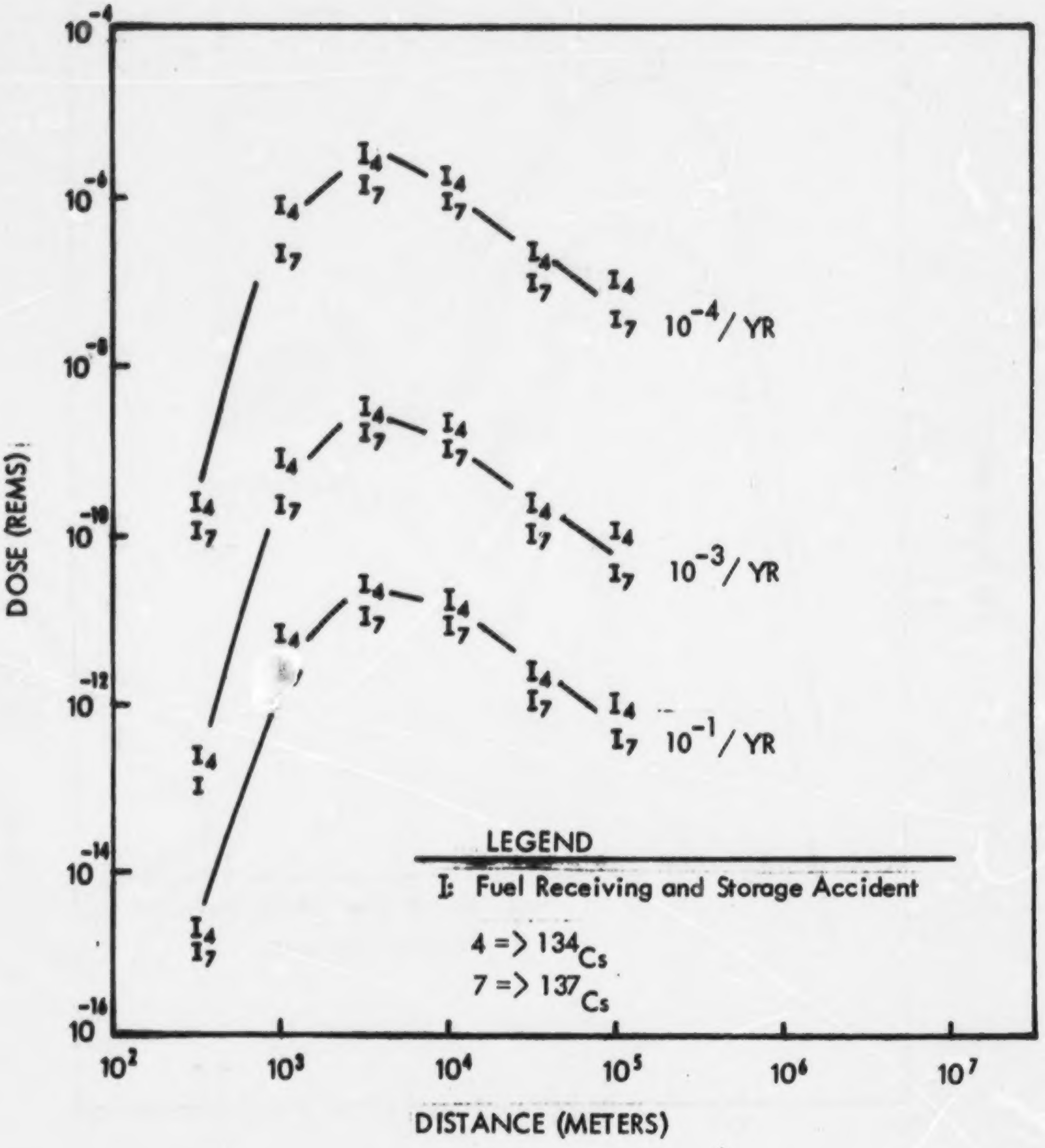

Figure VI-7 CESIUM DOSE AT DISTANCE FOR SEVERAL LIKELIHOODS

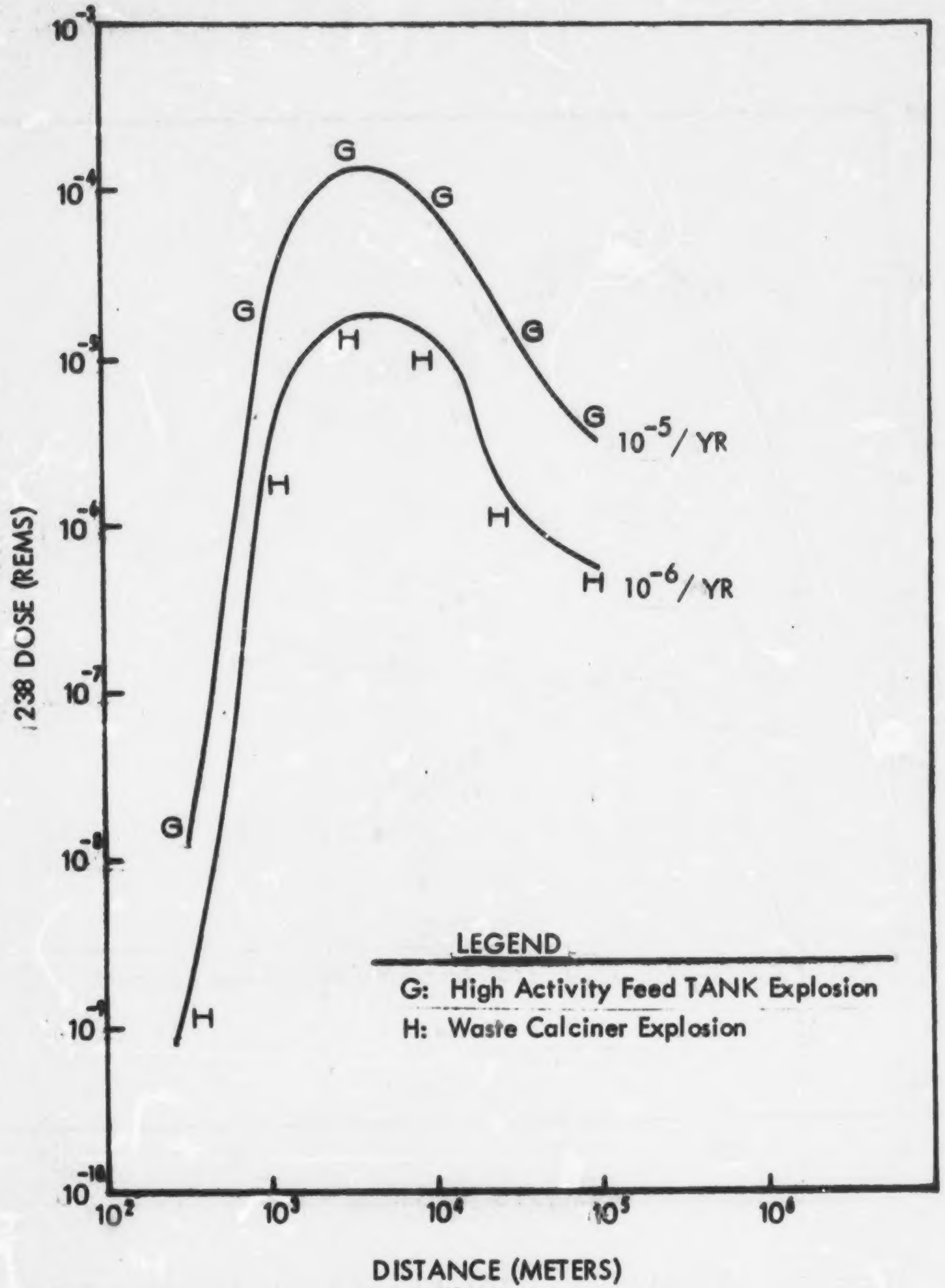

Figure VI-8 LUNG DOSE FROM PLUTONIUM VS DISTANCE FOR SEVERAL LIKELIHOODS

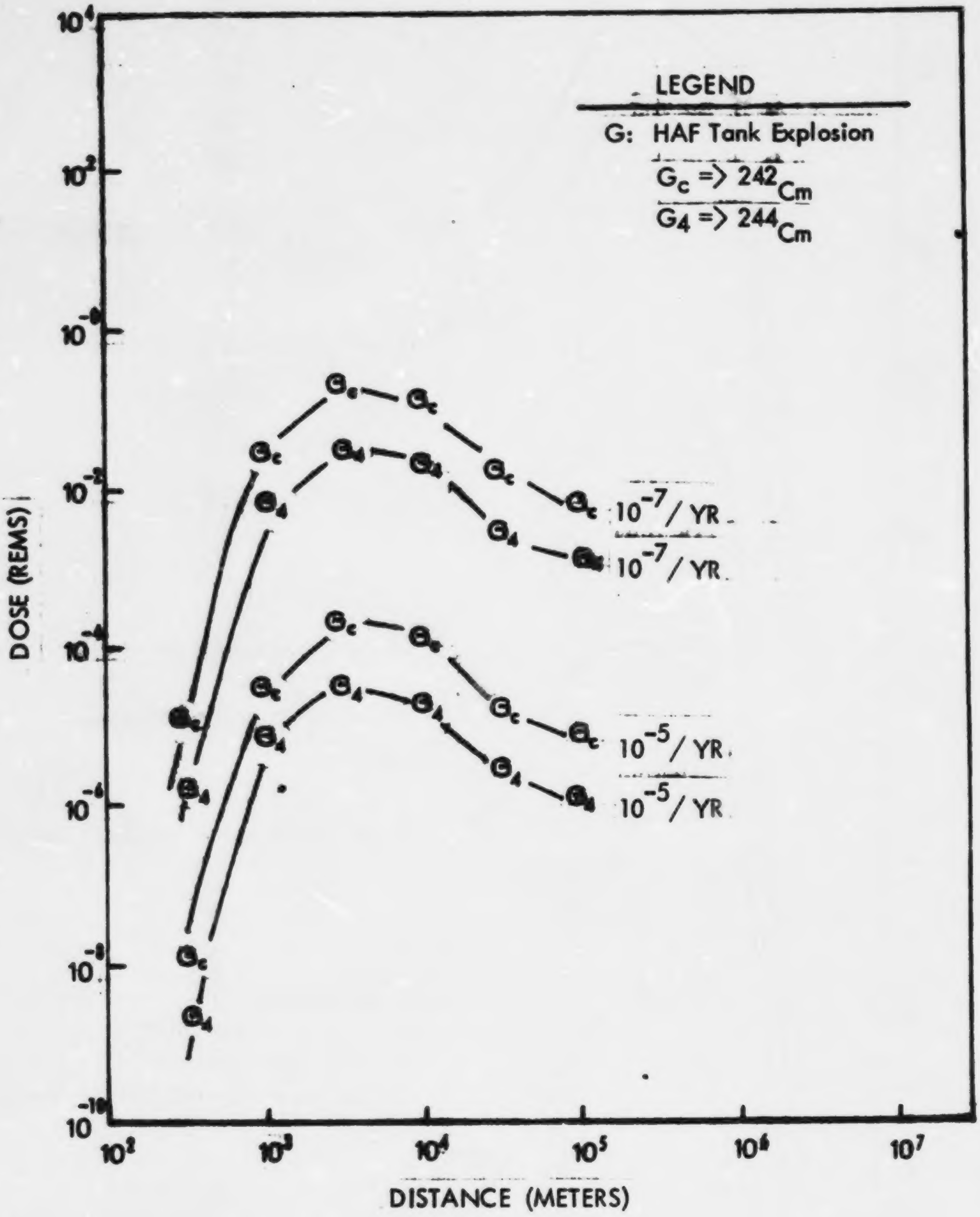

Figure VI-9 CURIUM DOSE AT DISTANCE FOR SEVERAL LIKELIHOODS

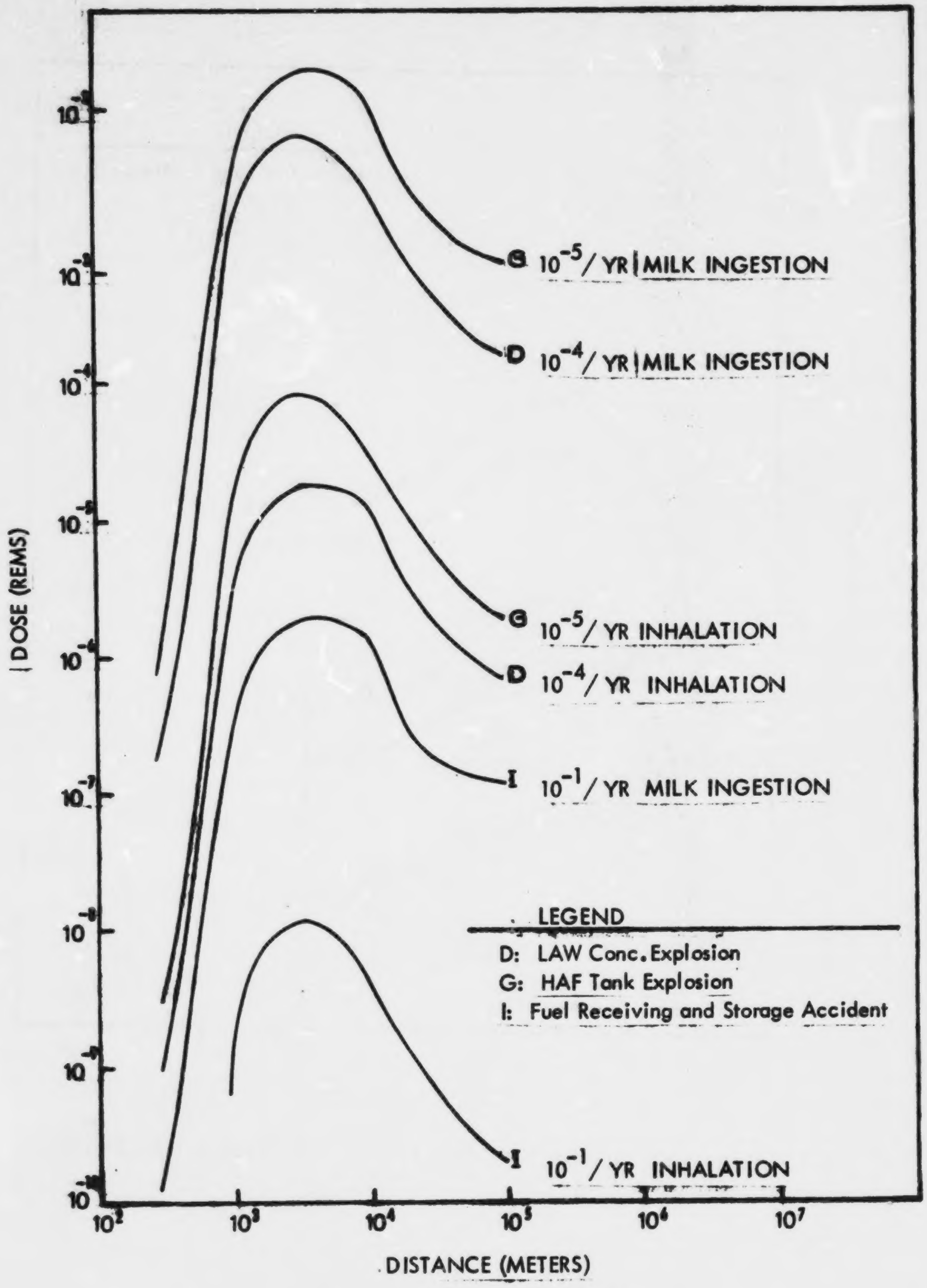

Figure VI-10a THYROID DOSE FROM IODINE-129 VS DISTANCE

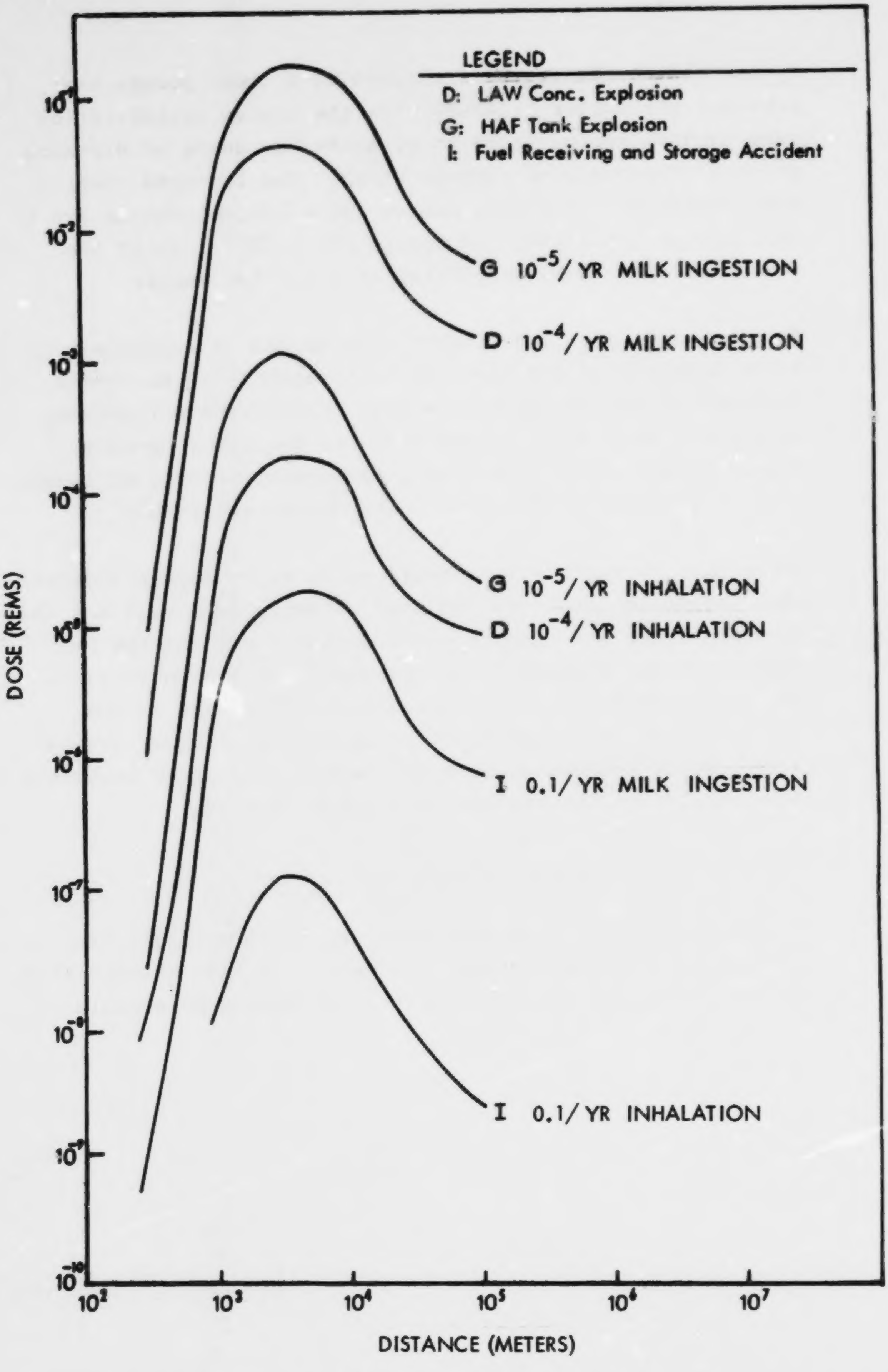

Figure VI-10b THYROID DOSE FROM IODINE-131 VS DISTANCE

By inspecting Figures VI-1 through VI-5 those points that provided the larger releases with the higher probabilities were used as source terms to generate the doses at distance given in Figures VI-6 through VI-10. The isotopes plotted were ruthenium 103 & 106, cesium 134 & 137, plutonium 238 & 241, curium 242 & 244, and iodine 129 & 131. Any of the other isotopes could be plotted in a similar manner.

The distance where the largest dose occurs is approximately 5,000 meters from the plant and this appears to be common for each of the isotopes examined. Category D meteorology is perhaps that which occurs with the highest frequency. A more pessimistic, less likely meteorology could, of course, lead to a series of different dose/distance curves.

Inspection of the results presented in this chapter indicate that incidents involving the high activity feed tank and the fuel receiving and storage cask, dominate much of the release data for non-plutonium releases. Plutonium releases are most evident in ion exchange resin fires and in the high activity waste concentrator and waste calciner explosions. Other incidents, such as low activity waste concentration explosions, contribute to a lesser extent.

3. Site Related Events

It is difficult in a generic study to utilize actual site related data in the accident analysis. We have assumed that the generic plant would be built to withstand forseeable site related events. Listed in Table VI-4 are several severe phenomena whose occurrence might damage a portion of the plant. While we cannot do a specific failure analysis for those initiating events, the following statements appear appropriate.

Were a severe tornado to strike a reprocessing plant it might initiate the following failures:

a. loss of offsite electric power
b. filter failure
c. missile penetration of a portion of the building
d. stack structural failure, either partial or complete
e. possible loss of storage pool water.

There appears to be practically no way in which a tornado can cause the process cell walls to fail. The loss of electric power or storage pool water are expected to be temporary in nature. The possible release of excess radioactivity due to filter failure or stack collapse will be counteracted by the extensive turbulence and dispersion caused by the tornado itself. Hence little radiological risk is expected from tornado induced events.

An earthquake can cause the following malfunctions:

a. possible structural failure
b. loss of offsite electric power
c. internal piping failures
d. stack collapse
e. loss of pool water.

TABLE VI-4

Selected Natural Event Data

Event	Frequency of Occurrence[19,57,58]
Tornado[19]	$6(10^{-4})$/yr.
Earthquake[19] Intensity IX	10^{-5}/yr.
Meteorites[57]	10^{-9}/yr.
Airplane Impact[58]	10^{-6}/yr.

The incident of importance here is the piping failure possibility. This event could cause releases to occur in many of the cells simultaneously. Building collapse is not expected to occur even during severe earth tremors. The releases would probably be liquid in nature and consequently would not contribute significantly to airborne releases. Moreover the frequency of 10^{-5}/year is about the same as the process cell incidents. Hence the earthquake induced releases will not impact the upper limit process cell accidents to any noticeable extent.

Airplane or meteorite impact with subsequent fire would cause local process cell failure. However these likelihoods are smaller than cell initiated events and are therefore not significant.

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APPENDIX A

Summary Table Describing the Basic Operations, Process Functions and Chemical Reactions in the Generic Spent Fuel Reprocessing Plant (Modeled after BNFP).

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PROCESS STEP	FUNCTION AND PRINCIPAL CHEMICAL REACTIONS	DESCRIPTION
Cask receiving and handling	Receipt and preparation of shipping cask for unloading	Cask and carrier will be monitored for outside contamination and washed to remove outside dirt. The cask will be removed and the condition of fuel and coolant determined by temperature, pressure, and coolant radioactivity measurements. The cask will be vented to the vessel off-gas system and the primary coolant replaced, if necessary. The cask will be placed in the cask unloading pool where the lid will be removed and fuel elements unloaded remotely under water shield. Empty casks will be decontaminated, monitored, and returned to customer.
Fuel storage and transfer	Storage of fuel elements until dissolution	Fuel element identity will be confirmed and the elements placed in storage canisters in the storage pool. Pool water will be circulated through heat exchangers, inorganic ion exchange beds, and filters to remove fuel decay heat and radioactive contaminants. Elements will be remotely transferred from the pool to the feed mechanism of shear.
Shear	Preparation of fuel for dissolution	Fuel elements will be mechanically chopped into small segments, exposing oxide fuel inside the elements to dissolution while outside cladding (stainless steel or zircaloy) will remain undissolved.

PROCESS STEP	FUNCTION AND PRINCIPAL CHEMICAL REACTIONS	DESCRIPTION
Dissolution and feed preparation	Conversion of the fuel to a liquid solution	The chopped fuel elements will be contacted with hot, concentrated nitric acid which will convert uranium, plutonium, and most of the fission products to soluble nitrate salts. Undissolved cladding (hulls) will remain in dissolver basket. Gases generated during dissolution will be channeled to off-gas treatment system. Nitrate salt solution will be transferred to tanks for sampling measurement and final acid adjustment.
Dissolution	$3UO_2 + 8HNO_3 \xrightarrow{<8M} 3UO_2(NO_3)_2 + 4H_2O + 2NO$ $UO_2 + 4HNO_3 \xrightarrow{>8M} UO_2(NO_3)_2 + 2H_2O + 2NO_2$ FISSION PRODUCTS + x $HNO_3 \rightarrow$ F.P. $(NO_3)_x + yH_2O + z\,NO$ $PuO_2 + 4HNO_3 \longrightarrow Pu(NO_3)_4 + 2H_2O$ $NO + 2HNO_3 \xrightarrow{>8M} 3NO_2 + H_2O$	
Solids handling and waste	Disposal of undissolved cladding hulls	The cladding hulls will be rinsed and transferred by shielded trailer to a burial ground. Intermittently, or in case of abnormalities during dissolution, batches of hulls will be checked for complete dissolution of plutonium and uranium.
Co-decontamination and partition cycle	Separation of the plutonium and uranium from the bulk of the fission products and partitioning of the plutonium from the uranium	Adjusted aqueous feed solution and tributyl phosphate (TBP) diluted in a normal paraffin hydrocarbon will be mixed counter-currently in a bank of centrifugal contactors. The organic solution, which preferentially extracts the

PROCESS STEP	FUNCTION AND PRINCIPAL CHEMICAL REACTIONS	DESCRIPTION
Extraction	$UO_2^{++} + 2NO_3^- + 2TBP \xrightarrow{n\text{-}C_{12}H_{26}} UO_2(NO_3)_2 \cdot 2TBP$ $Pu^{+4} + 4NO_3^- + 2TBP \xrightarrow{n\text{-}C_{12}H_{26}} Pu(NO_3)_4 \cdot 2TBP$ $PuO_2^{++} + 2NO_3^- + 2TBP \xrightarrow{n\text{-}C_{12}H_{26}} PuO_2(NO_3)_2 \cdot 2TBP$	nitrate complexes of tetravalent plutonium and hexavalent uranium, will exit from the centrifugal contactor and **pass** through a pulsed scrub column where an aqueous nitric acid solution will remove extracted fission products from the organic stream. The organic stream will pass through a partitioning column where plutonium will be reduced to the inextractable trivalent state and stripped into another aqueous nitric acid stream containing hydrazine. The organic stream will pass through another column where the uranium will be stripped into acidified water. **(Alternatively, anion exchange would be used for partitioning plutonium and uranium into separate streams)**
Reduction and Partitioning	$Pu^{+4}_{aq} + 1e^- \longrightarrow Pu^{+3}_{aq}$ $UO_2{}^{++}_{aq} + 2e^- + 4H^+ \longrightarrow U^{+4}_{aq} + 2H_2O$ $U^{+4}_{aq} + 2Pu^{+4}_{aq} + 2H_2O \longrightarrow UO_2^{++} + 2Pu^{+3} + 4H^+$ $N_2H_4 + 2HNO_2 \longrightarrow N_2 + N_2O + 3H_2O$	
Stripping	$UO_2(NO_3)_2 \cdot 2TBP + H_2O + 2H^+ \rightarrow UO_2{}^{++}_{aq} + 2HNO_3 + H_2O + 2TBP$	
Second uranium cycle	Further decontamination of uranium from fission products	Nitric acid will be added to the aqueous strip stream containing the uranium, and the uranyl nitrate complex will again be preferentially extracted by another TBP solution in a pulsed column. Before leaving the column,

PROCESS STEP	FUNCTION AND PRINCIPAL CHEMICAL REACTIONS	DESCRIPTION
		the organic stream will be scrubbed successively with strong and dilute nitric acid solutions which remove extracted ruthenium and zirconium-niobium, respectively. Uranium will be stripped from the organic stream in another column, using acidified water, and this solution will be subsequently concentrated by evaporation.
Uranium silica gel, product storage	Final decontamination and disposition of uranium	Concentrated uranium solution will be passed through silica gel beds to remove traces of zirconium-niobium. Uranyl nitrate product solution will be analyzed and stored in tanks until shipment.
Second and third plutonium cycle, storage, and shipping	Final decontamination and disposition of plutonium	Plutonium in aqueous stream leaving partitioning column will be reoxidized to the extractable tetravalent state, which will be preferentially extracted into the TBP organic stream in a pulsed column. In the same column, the organic stream will be scrubbed successively with strong and dilute nitric acid solutions, which will remove extracted ruthenium and zirconium-niobium, respectively. The organic stream will pass through a strip column where plutonium will be reduced to inextractable trivalent state, which will transfer to the aqueous stream of dilute nitric acid and hydrazine. The extraction-stripping
Oxidation	$Pu^{+3} + 2NO_2 + H^+ \longrightarrow$ $Pu^{+4} + HNO_3 + NO$	
Extraction	$Pu^{+4} + 4NO_3^- + 2TBP \xrightarrow{n-C_{12}H_{26}}$ $Pu(NO_3)_4 \cdot 2TBP$	
Reduction and Stripping	$Pu^{+4}_{aq} + 1e^- \longrightarrow Pu^{+3}_{aq}$ $UO_2{}^{++}_{aq} + 2e^- + 4H^+ \longrightarrow$ $U^{+4}_{aq} + 2H_2O$	

PROCESS STEP	FUNCTION AND PRINCIPAL CHEMICAL REACTIONS	DESCRIPTION
	$U^{+4}_{aq} + 2Pu^{+4}_{aq} + 2H_2O \longrightarrow$ $UO_2^{++} + 2Pu^{+3} + 4H^+$	sequence will be repeated in the third plutonium cycle for further decontamination. A TBP scrub stream will
Scrubbing	$TBP + UO_2^{++} + 2NO_3^- \xrightarrow{n-C_{12}H_{26}}$ $UO_2(NO_3)_2 \cdot 2TBP$	remove residual uranium from the plutonium aqueous stream as it leaves the last strip column. Plutonium concentration will be accomplished by maintaining a high ratio of organic to aqueous flow in the strip columns. Final plutonium nitrate solution will be washed with an organic stream of normal paraffin hydrocarbon (diluent for TBP) to remove traces of TBP and phosphate. Product solution will be analyzed and stored in tanks until shipment. Solvent streams leaving plutonium cycles will pass through a strip column to remove residual inextractable species of uranium and plutonium and will be recycled to the co-decontamination cycle.
No. 1 solvent system Carbonate wash	Removal of degradation products from solvent $Na_2CO_3 + 2(C_4H_9)_2HPO_4 \longrightarrow$ $2(C_4H_9)_2NaPO_4 + H_2CO_3$ $2RCH_2NO_2 \rightleftharpoons 2RCH = NOOH +$ $Na_2CO_3 \longrightarrow 2RCH = NOONa +$ H_2CO_3	Organic solvent stream from co-decontamination and partition cycle will be washed successively with dilute aqueous solutions of sodium carbonate, nitric acid, and sodium carbonate (or sodium hydroxide) to remove organic degradation products by extraction or precipitation; precipitated solids will be removed by a filter. Fresh TBP or diluent (normal paraffin hydrocarbon) will be added,

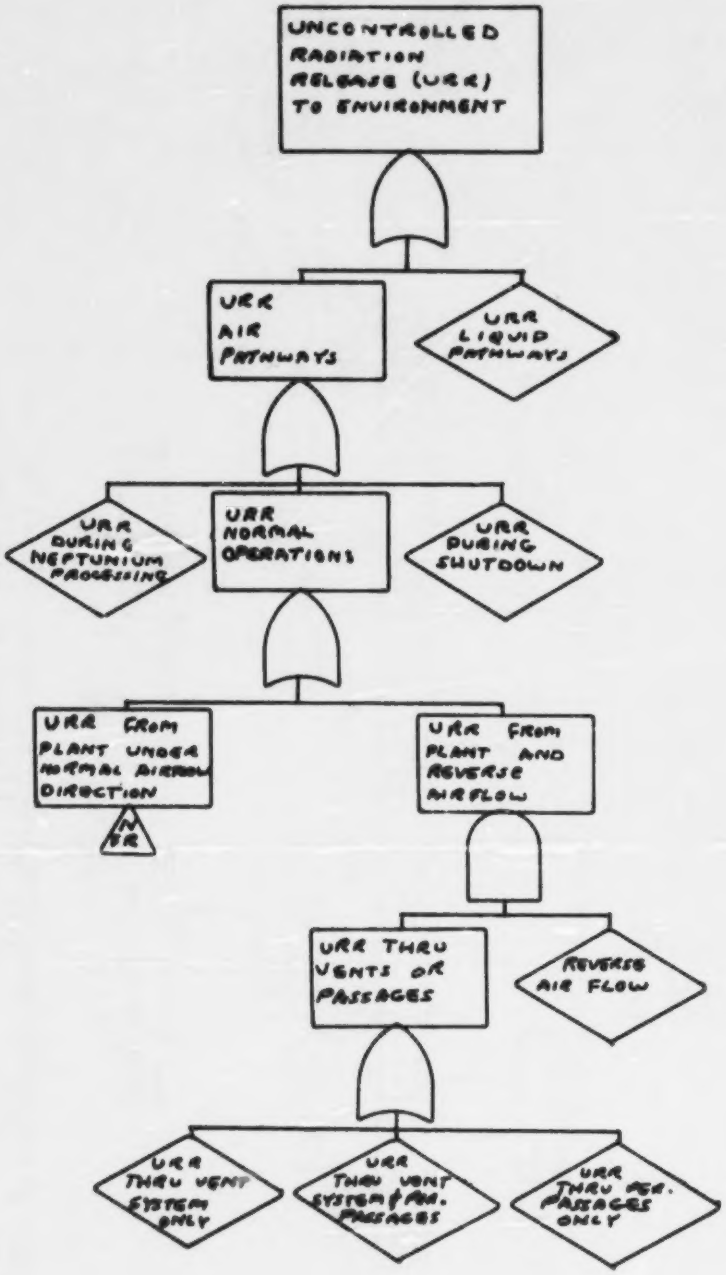
PROCESS STEP	FUNCTION AND PRINCIPAL CHEMICAL REACTIONS	DESCRIPTION
		as required, to maintain proper TBP concentration or total solvent inventory.
No. 2 solvent system	Removal of degradation products from solvent	Organic solvent stream from second uranium cycle will be treated similarly to No. 1 system, except the second alkaline wash will be omitted.
Liquid waste treating and storage	Disposal of liquid waste streams with minimum residual waste volume for storage.	The highly radioactive waste stream from the co-decontamination cycle will be concentrated by evaporation; acidity of the concentrated bottoms will be reduced to permit long-term storage in stainless steel tanks by reacting with a sugar solution; overheads will be fed to the low-activity evaporator for further decontamination. Most of the remaining nitric acid waste streams containing low levels of fission products, uranium, and plutonium will be concentrated in the low-activity waste evaporator; concentrated bottoms will be recycled to the co-decontamination cycle; overheads will be condensed and fed to the acid recovery system. Miscellaneous waste streams, containing salts, low levels of fission products and no appreciable uranium or plutonium, will be acidified and concentrated in the general-purpose evaporator; bottoms will be stored; overheads will be monitored for radioactivity content and then
Acid reduction	$C_{12}H_{22}O_{11} + 18.2\ HNO_3 \longrightarrow 12CO_2 + 14.9\ NO + 3.3\ NO_2 + 20.1\ H_2O$	

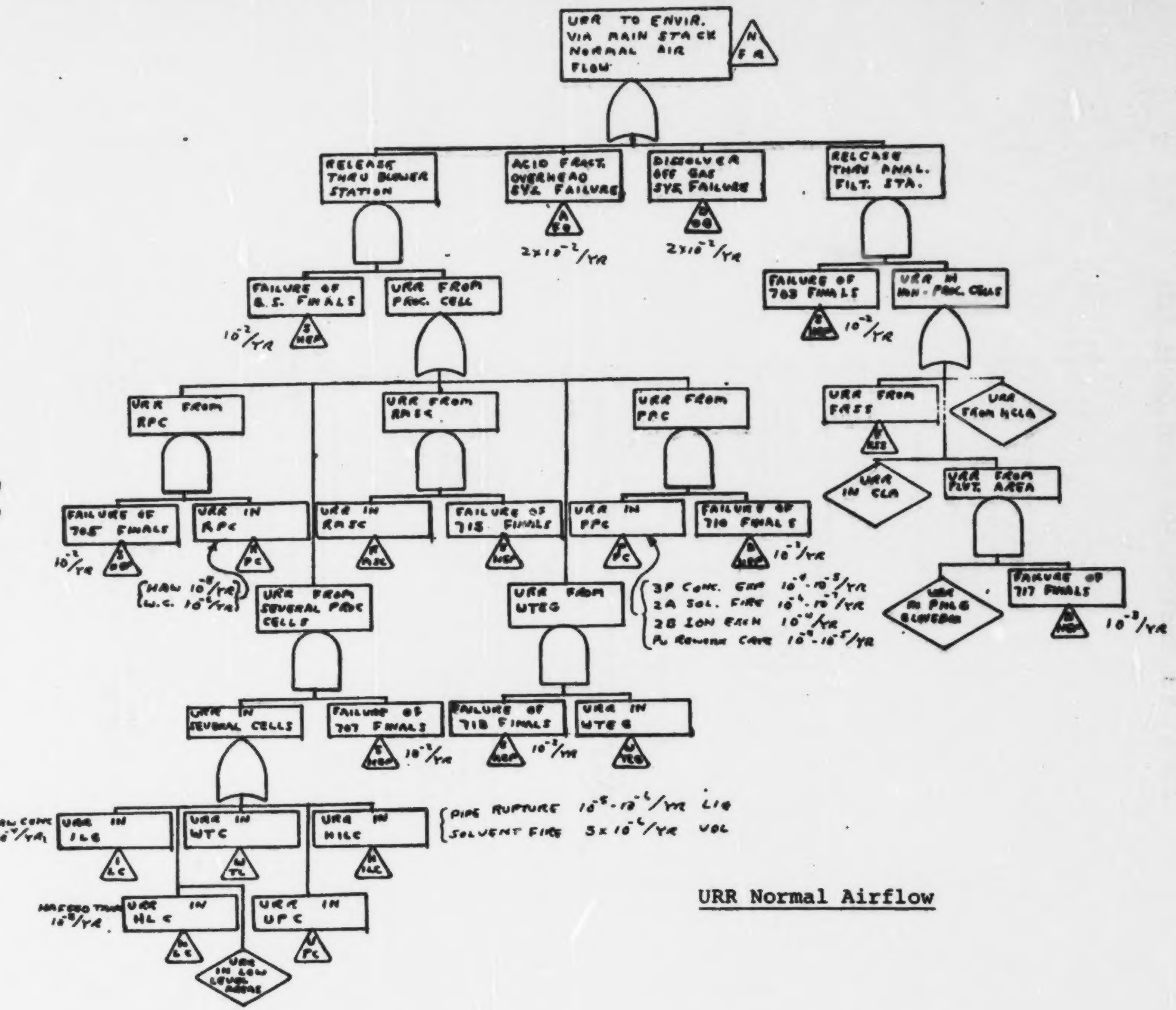
PROCESS STEP	FUNCTION AND PRINCIPAL CHEMICAL REACTIONS	DESCRIPTION
		discharged.
Nitric acid recovery and storage	Recovery of nitric acid and reduction of nitrogen oxides release to the environs.	Overheads from LAW evaporator will contain most of the tritium (as tritiated water) and some undestroyed nitric acid from the process; they will be condensed and fed to the fractionator which concentrates nitric acid. Recovered acid will be stored and used in make-up of various acid streams; overheads containing tritiated water will be monitored for radioactivity (other than tritium) and released to the stack.
Off-gas treating	Removal of radioactive and other pollutants from gaseous effluents	Off-gas from dissolver will pass through a scrubber where radioactive iodine will be removed by contact with dilute aqueous solution of nitric acid and mercurous/mercuric nitrate; it will subsequently pass through an acid absorber where nitrogen oxides will be removed. Dissolver off-gas and vessel off-gas streams will be combined, passed through another mercurous/mercuric nitrate scrubber, an iodine adsorber bed, and a high-efficiency filter before release to the stack.
Iodine Scrub	*$Hg(NO_3)_2 + 4I^- \longrightarrow HgI_4^{=} + 2NO_3^-$ *$Hg(NO_3)_2 + 3I^- \longrightarrow HgI_3^- + 2NO_3^-$ *Hg/I_2 mole ratio $\geqslant 4$	

Appendix B: Fault Trees Used in Risk Assessment

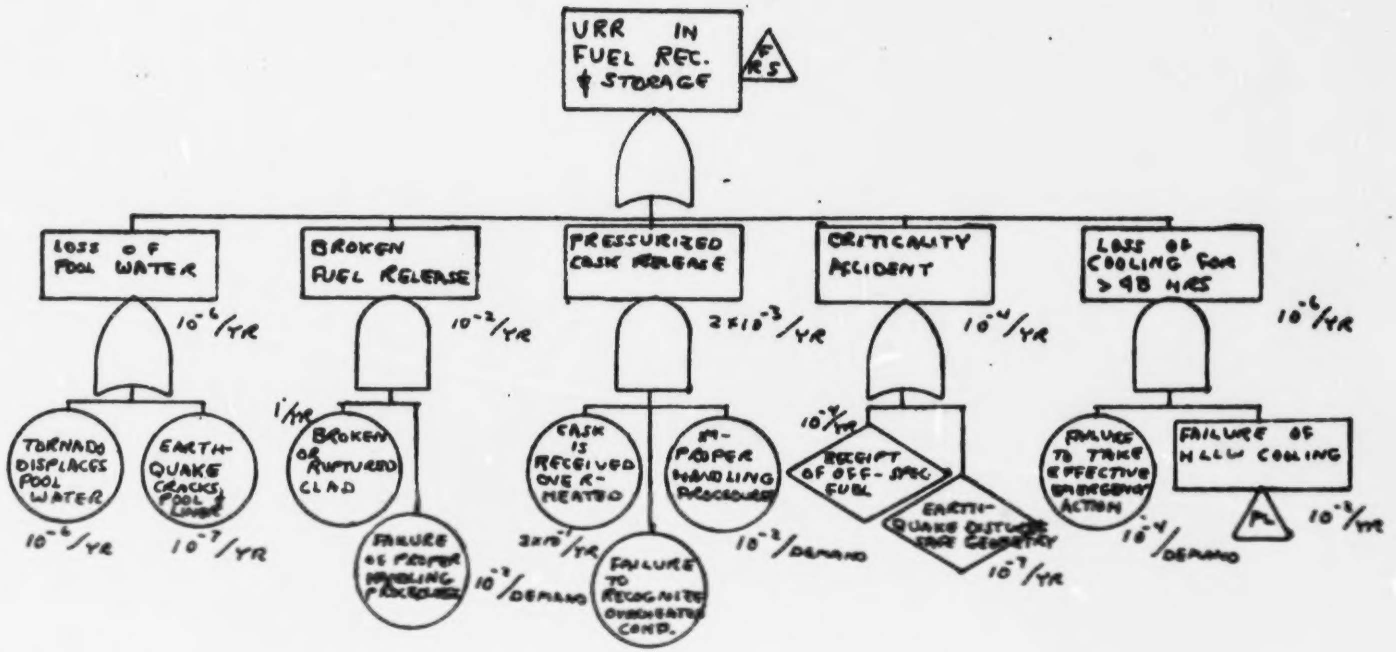
Page	Description	Abbreviation
B-1	Top of the Reprocessing Plant Fault Tree	---
B-2	URR Normal Airflow	NFR
B-3	URR in Fuel Receiving and Storage	FRS
B-4	URR in Remote Process Cell	RPC
B-5	URR in High-Level Cell	HLC
B-6	URR in High Intermediate-Level Cell	HILC
B-7	URR in Intermediate Level Cell	ILC
B-8	URR in Plutonium Process Cell	PPC
B-9	URR in Plutonium Loadout Cell	PLC
B-10	URR from High-Level Liquid Waste Tank	LPS,WTV,WTS
B-11	Failure of Heat Transfer from HLLW	HT
B-12	Failure to Institute Emergency Reflood	ER
	Dissolver Off-Gas System	DOG
	Vessel Off-Gas System	VOG
B-13	Cooling Water Failure	WS
	Dissolver and Vessel Off-Gas Common System-	VD
B-14	URR from Resin Reaction	RRR
B-15	Solvent Fire in Partition and Purification-	SF
B-16	Red Oil Explosion	RO
B-17	Steam Explosion	S
B-18	Criticality Accident in Process Cell	CP
B-19	Controller Fails Unsafe	C
	Failure of Personnel Intervention	PI
	On-Site Power Failure	OSP
B-20	Acid Fraction Overhead	AFO
	Pump Failure	P

Top of the Reprocessing Plant Fault Tree


UNCONTROLLED RADIATION RELEASE (URR) TO ENVIRONMENT
URR AIR PATHWAYS
URR LIQUID PATHWAYS
URR DURING NEPTUNIUM PROCESSING
URR NORMAL OPERATIONS
URR DURING SHUTDOWN
URR FROM PLANT UNDER NORMAL AIRFLOW DIRECTION
N FR
URR FROM PLANT AND REVERSE AIRFLOW
URR THRU VENTS OR PASSAGES
REVERSE AIR FLOW
URR THRU VENT SYSTEM ONLY
URR THRU VENT SYSTEM & PER. PASSAGES
URR THRU PER. PASSAGES ONLY


URR TO ENVIR. VIA MAIN STACK NORMAL AIR FLOW
N FR
RELEASE THRU BLOWER STATION
ACID FRACT. OVERHEAD SYS. FAILURE
2x10⁻²/YR
DISSOLVER OFF GAS SYS. FAILURE
2x10⁻²/YR
RELEASE THRU ANAL. FILT. STA.
FAILURE OF B.S. FINALS
10⁻²/YR
URR FROM PROC. CELL
FAILURE OF 703 FINALS
10⁻²/YR
URR IN NON-PROC. CELLS
URR FROM RPC
URR FROM RMSC
URR FROM PPC
URR FROM FRSS
URR FROM HCLA
URR IN CLA
URR FROM PLUT. AREA
FAILURE OF 705 FINALS
10⁻²/YR
URR IN RPC
HAW 10⁻⁸/YR
W.C. 10⁻⁶/YR
URR IN RMSC
FAILURE OF 715 FINALS
URR IN PPC
FAILURE OF 710 FINALS
10⁻³/YR
3P CONC. EXP 10⁻⁴-10⁻⁵/YR
2A SOL. FIRE 10⁻⁶-10⁻⁷/YR
2B ION EXCH 10⁻⁴/YR
Pu REWORK CAVE 10⁻⁴-10⁻⁵/YR
URR IN PNLG GLOVEBOX
FAILURE OF 717 FINALS
10⁻³/YR
URR FROM SEVERAL PROC CELLS
URR FROM UTEG
URR IN SEVERAL CELLS
FAILURE OF 707 FINALS
10⁻³/YR
FAILURE OF 718 FINALS
10⁻³/YR
URR IN UTEG
LAW CONC 10⁻⁴/YR
URR IN ILC
URR IN WTC
URR IN HILC
PIPE RUPTURE 10⁻⁵-10⁻⁶/YR LIQ
SOLVENT FIRE 5x10⁻⁶/YR VOL
HAFEED TANK 10⁻⁵/YR
URR IN HLC
URR IN UPC
URR IN LOW LEVEL AREAS

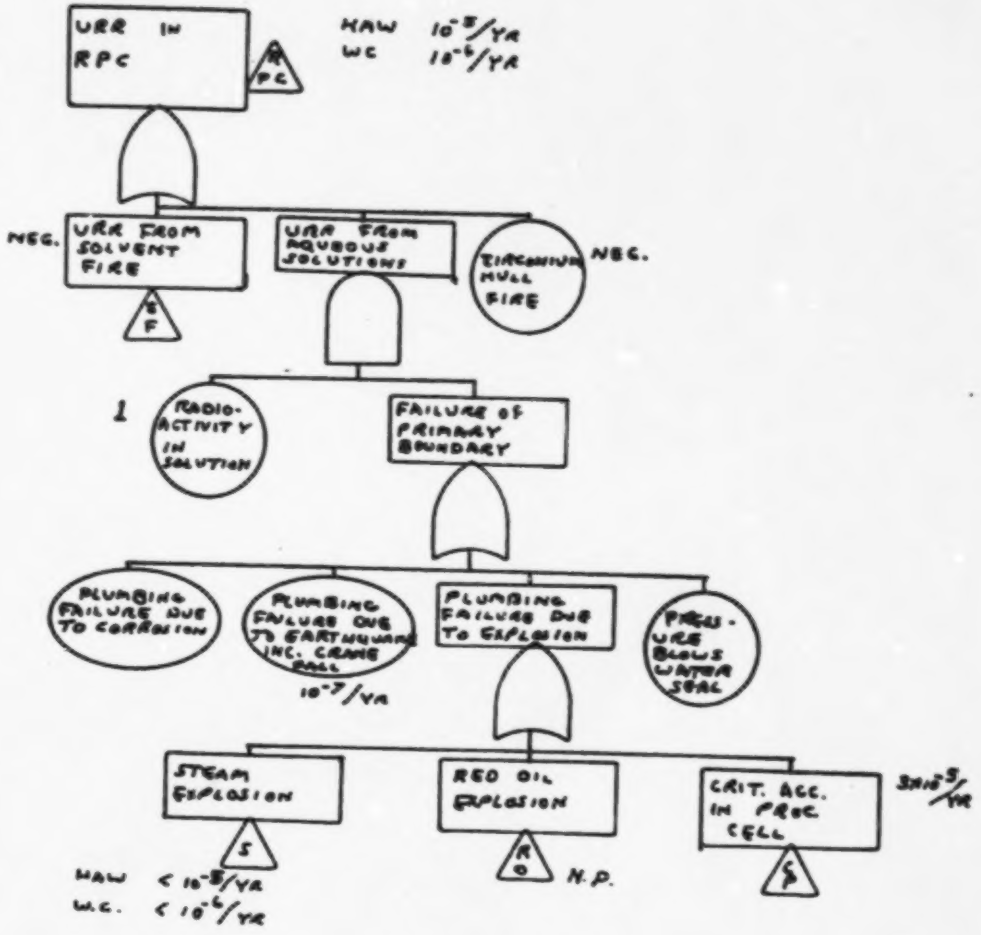
URR Normal Airflow


URR IN FUEL REC. ¢ STORAGE
F RS
LOSS OF POOL WATER
10^-6/YR
TORNADO DISPLACES POOL WATER
10^-6/YR
EARTH-QUAKE CRACKS POOL ¢ LINER
10^-7/YR
BROKEN FUEL RELEASE
10^-2/YR
1/YR
BROKEN OR RUPTURED CLAD
FAILURE OF PROPER HANDLING PROCEDURES
10^-2/DEMAND
PRESSURIZED CASK RELEASE
2×10^-3/YR
CASK IS RECEIVED OVER-HEATED
2×10^-1/YR
FAILURE TO RECOGNIZE OVERHEATED COND.
IM-PROPER HANDLING PROCEDURES
10^-2/DEMAND
CRITICALITY ACCIDENT
10^-4/YR
10^-4/YR
RECEIPT OF OFF-SPEC FUEL
EARTH-QUAKE DISTURBS SAFE GEOMETRY
10^-7/YR
LOSS OF COOLING FOR >48 HRS
10^-6/YR
FAILURE TO TAKE EFFECTIVE EMERGENCY ACTION
10^-4/DEMAND
FAILURE OF HLLW COOLING
FL
10^-2/YR

URR in Fuel Receiving and Storage

LIST OF EQUIPMENT

15-C-101	#1 DISSOLVER
15-C-102	#2 DISSOLVER
15-C-103	#3 DISSOLVER
41-D-444	HWW HEAD POT
15-E-132	#1 DISSOLVER HEATER/COOLER
15-E-133	#2 DISSOLVER HEATER/COOLER
15-E-134	#3 DISSOLVER HEATER/COOLER
41-E-401	HAW CONCENTRATOR REBOILER
41-E-400	HAW CONCENTRATOR
15-E-154	#1 DISSOLVER SCREEN POT
15-E-155	#2 DISSOLVER SCREEN POT
15-E-156	#3 DISSOLVER SCREEN POT
06-R-001A/B	MAINTENANCE CEILING PLUG
06-R-001C/D	MAINTENANCE CEILING PLUG
90-R-001	SHIELDING DOOR
06-T-008	FUEL TRANSFER CONVEYOR
06-T-016	FUEL TRANSFER CONVEYOR DRIVE
13-T-101	TRANSVERSE FUEL PUSHER
12-T-102	MAINTENANCE CRANE 35-TON
12-T-103	BASKET HANDLING CRANE
12-T-104	POWER MANIPULATOR
12-T-106A/D	DISSOLVER BASKET STORAGE RACK
13-T-107	FUEL TRANSFER TABLE
46-T-401	DISSOLVER BASKET DUMPER
46-T-403	HULL MONITOR
46-T-411	SUMP JET BASE
46-T-412	SUMP JET BASE
06-T-043	PASS THROUGH PORT
07-V-004	SHIELDING WINDOW
07-V-005	SHIELDING WINDOW
07-V-006	SHIELDING WINDOW
07-V-026	SHIELDING WINDOW
07-V-010A/E	PERISCOPE SLEEVE & PLUG
07-V-010B/F	DITTO
07-V-010C/G	DITTO
07-V-010D/H	DITTO
07-V-021	SHIELDING WINDOW
14-T-101	SHEAR
14-T-104	SHEAR SUPPORT FRAME
14-T-105	SHEAR FEED MAGAZINE
14-T-106	SHEAR MAG. TRANSITION SECT.
14-T-107	SHEAR FEED CHAIN & MAGAZINE
14-T-108	SHEAR FEED RAM DRIVE UNIT
16-T-152	DIVERTER
16-T-153	DIVERTER DRIVE UNIT
18-Z-101	FUEL TRANSFER HATCH
	WASTE CALCINER WC



URR IN RPC
R PC
HAW 10^{-5}/YR
WC 10^{-6}/YR
NEG.
URR FROM SOLVENT FIRE
S F
URR FROM AQUEOUS SOLUTIONS
ZIRCONIUM HULL FIRE
NEG.
1
RADIO-ACTIVITY IN SOLUTION
FAILURE OF PRIMARY BOUNDARY
PLUMBING FAILURE DUE TO CORROSION
PLUMBING FAILURE DUE TO EARTHQUAKE INC. CRANE FALL
10^{-7}/YR
PLUMBING FAILURE DUE TO EXPLOSION
PRESS-URE BLOWS WATER SEAL
STEAM EXPLOSION
S
HAW $< 10^{-5}$/YR
W.C. $< 10^{-6}$/YR
RED OIL EXPLOSION
R O
N.P.
CRIT. ACC. IN PROC CELL
3×10^{-5}/YR
$

URR in Remote Process Cell

EQUIPMENT LIST

15-D-107 NO. 1 DISSOLVER TRANSFER TANK
15-D-108 NO 2 " " "
15 D 109 NO 3 " " "
15 D 117 NO 4 " " "
15 D 140 NO 1 " HEAD POT "
15 D 141 NO 2 " " " "
15 D 142 NO 3 " " " "
15 D 143 NO 4 " " " "

17 D 103 NO. 1 FEED ADJUSTMENT TANK
17 D 104 NO 2 " " "
17 D 111 DISSOLVER FLUSH ACCUMULATOR
17 D 113 FEED SURGE TANK
17 D 144 DISSOLVER PRODUCT HEAD POT

18 D 101 ACCOUNTABILITY TANK

21 D 201 HA FEED TANK
21 D 240 HAF CONSTANT LEVEL HEAD POT
22 D 263 2DW HEAD POT

41 D 404 HWW CATCH TANK
41 D 405 HWW SAMPLE TANK
41 D 417 HAW SURGE TANK
41 D 481 HAW HEAD POT

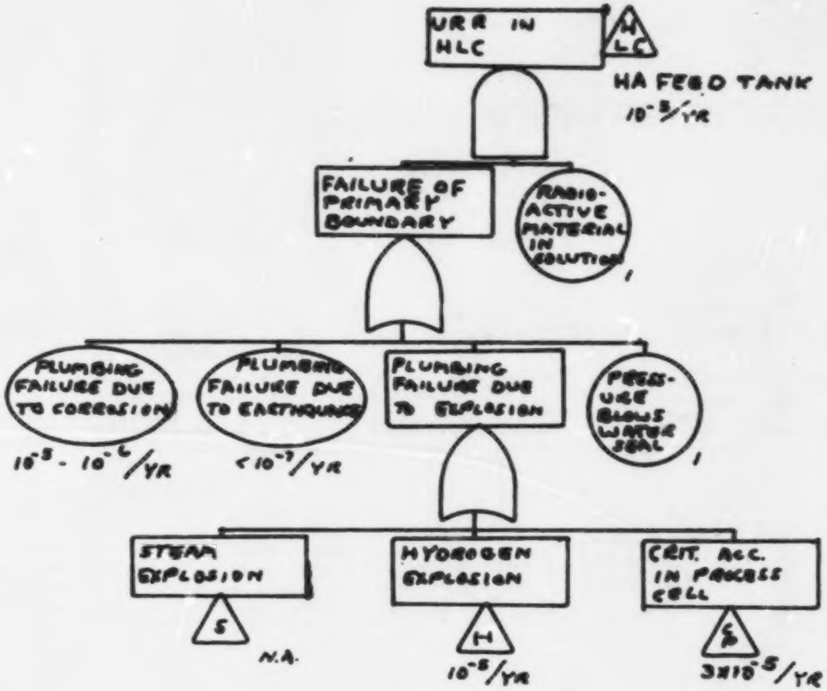
43 D 408 GENERAL PURPOSE CONCENTRATOR TANK
43 D 412 SUMP COLLECTION TANK
43 D 447 G P FEED HEAD POT

61 D 602 DISSOLVER ACID SURGE TANK
61 D 617 HULL RINSE SURGE TANK
61 D 655 HULL RINSE CONSTANT LEVEL HEAD POT
61 D 656 HULL RINSE MIX FEED HEAD POT
61 D 659 NO 1 DISSOLVER ACID HEAD POT
61 D 660 NO 2 " " " "
61 D 661 NO 3 " " " "
61 D 662 NO 4 " " " "
61 D 663 DISSOLVER HULL RINSE SEAL HEAD POT

VALVE POT NO	SAMPLE POINT	DESCRIPTION
80 D 001	N-01	ACCOUNTABILITY TANK
80 D 002	N-02	NO. 1 FEED ADJUSTMENT TANK
80 D 012	N-12	HAW SURGE TANK
80 D 013	N-13	LAW CONCENTRATOR FEED TANK
80 D 016	N-16	GEN PURPOSE CONCENTRATOR FEED TANK
80 D 019	N-19	HWW SAMPLE TANK
80 D 020	N-20	SUMP COLLECTION TANK
80 D 032	N-32	LAWB CHECK TANK
80 D 035	N-35	2DW LINE
80 D 061	N-61	DISSOLVER FLUSH ACCUMULATOR
80 D 062	N-62	NO. 2 FEED ADJUSTMENT TANK
80 D 070	N-70	NO 1 DISSOLVER TRANSFER TANK
80 D 071	N 71	NO. 2 " " "
80 D 072	N 72	NO. 3 " " "
80 D 073	N-73	NO. 4 " " "
80 D 074	N-74	FEED SURGE TANK

43 E 419 SUMP COLLECTION TANK REFLUX CONDENSER
43 E 421 GP FEED TANK REFLUX CONDENSER

16 A 33 CENTRIFUGE


URR IN HLC
HLC
HA FEED TANK
10^{-5}/YR
FAILURE OF PRIMARY BOUNDARY
RADIO-ACTIVE MATERIAL IN SOLUTION
1
PLUMBING FAILURE DUE TO CORROSION
10^{-5} - 10^{-6}/YR
PLUMBING FAILURE DUE TO EARTHQUAKE
$<10^{-7}$/YR
PLUMBING FAILURE DUE TO EXPLOSION
PRESSURE BLOWS WATER SEAL
1
STEAM EXPLOSION
S
N.A.
HYDROGEN EXPLOSION
H
10^{-5}/YR
CRIT. ACC. IN PROCESS CELL
CP
3×10^{-5}/YR

URR in High-Level Cell

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EQUIPMENT LIST

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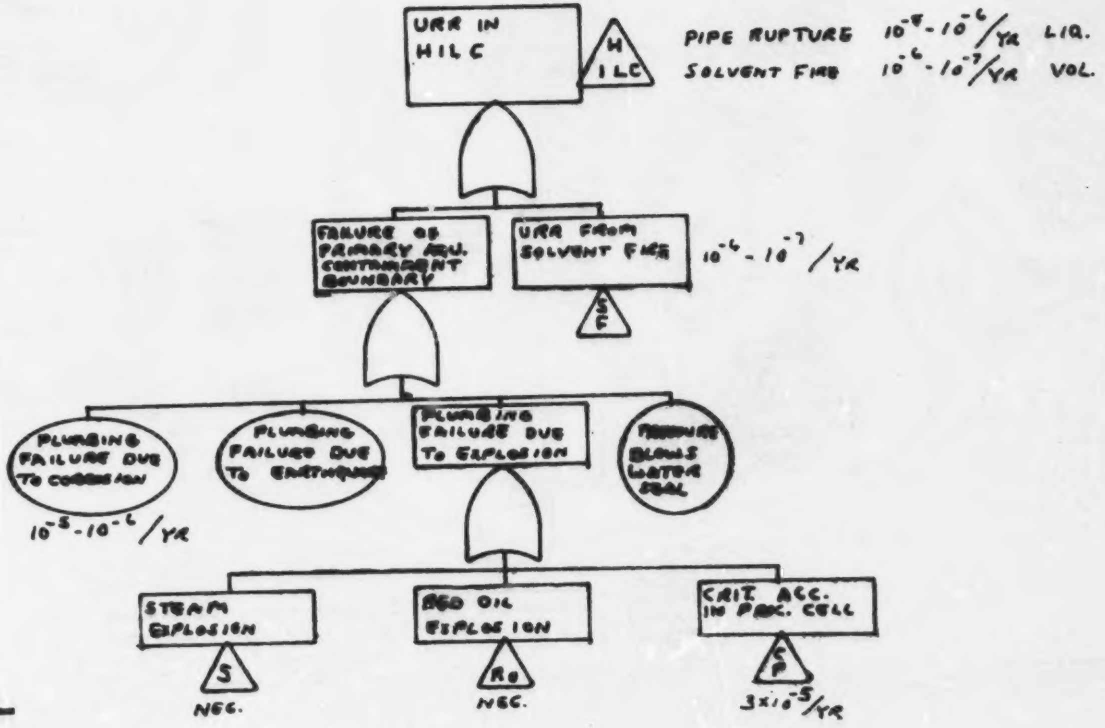
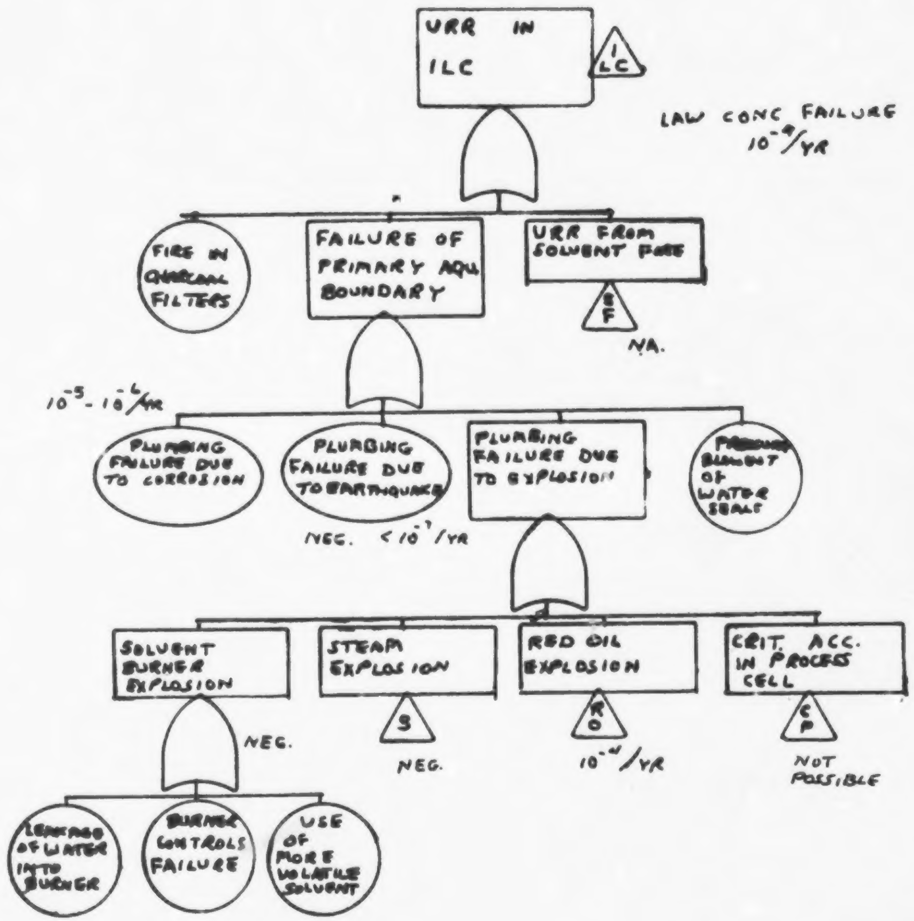

URR IN HILC
H ILC
PIPE RUPTURE $10^{-5}-10^{-6}$/YR LIQ.
SOLVENT FIRE $10^{-6}-10^{-7}$/YR VOL.
FAILURE OF PRIMARY AQU. CONTAINMENT BOUNDARY
URR FROM SOLVENT FIRE
$10^{-6}-10^{-7}$/YR
SF
PLUMBING FAILURE DUE TO CORROSION
$10^{-5}-10^{-6}$/YR
PLUMBING FAILURE DUE TO EARTHQUAKE
PLUMBING FAILURE DUE TO EXPLOSION
TORNADO BLOWS WATER SEAL
STEAM EXPLOSION
S
NEG.
RED OIL EXPLOSION
RO
NEG.
CRIT. ACC. IN PROC. CELL
CP
3×10^{-5}/YR

URR in High Intermediate-Level Cell

EQUIPMENT LIST

45-C-490	NO₂ ABSORBER
54-C-500	ACID FRACTIONATOR
42-C-401	SOLVENT BATCH STRIPPING TANK
42-C-402	SOLVENT BURNER FEED TANK
42-C-406	LAW CONCENTRATOR FEED TANK
42-C-434	LAWB CHECK TANK
42-C-440	SOLVENT BURNER QUENCH POT
42-C-445	42-D-406 HEAD POT
42-C-446	LAWB HEAD POT
42-D-46	42-C-440 HEAD POT
45-D-432	VACUUM BREAKER TANK
45-C-44	VESSEL OFF-GAS KO POT
45-[illegible]-472	NO₂ ABSORBER BOTTOMS HEAD POT
54-[illegible]-560	AF CHLORIDE REMOVAL HEAD POT
62-D-605	RECOVERED ACID STORAGE TANK
42-E-402	LAW CONCENTRATOR REBOILER
42-E-46	LAW CONCENTRATOR
45-E-4[illegible]5	VESSEL OFF-GAS CONDENSER
45-E-43	OFF-GAS HEATER (STEAM)
45-E-432	OFF-GAS HEATER (ELECT)
54-E-50[illegible]	ACID FRACTIONATOR REBOILER
54-E-5[illegible]5	ACID FRACTIONATOR CONDENSER
54-E-5[illegible]0	RECOVERED ACID COOLER
42-F-40[illegible]	WASTE SOLVENT BURNER
45-K-425	NO. 1 IODINE SCRUBBER
45-K-426	NO. 2 IODINE SCRUBBER
80-C-810	SAMPLER VALVE POT
80-C-81[illegible]	SAMPLER VALVE POT
80-D-817	SAMPLER VALVE POT
80-D-827	SAMPLER VALVE POT
80-C-847	SAMPLER VALVE POT


URR IN ILC
I LC
LAW CONC FAILURE 10^{-9}/YR
FIRE IN CHARCOAL FILTERS
FAILURE OF PRIMARY AQU BOUNDARY
URR FROM SOLVENT FIRE
S F
NA.
10^{-5} - 10^{-6}/YR
PLUMBING FAILURE DUE TO CORROSION
PLUMBING FAILURE DUE TO EARTHQUAKE
NEG. < 10^{-7}/YR
PLUMBING FAILURE DUE TO EXPLOSION
PREMATURE BLOWOUT OF WATER SEALS
SOLVENT BURNER EXPLOSION
NEG.
STEAM EXPLOSION
S
NEG.
RED OIL EXPLOSION
R O
10^{-4}/YR
CRIT. ACC. IN PROCESS CELL
C P
NOT POSSIBLE
LEAKAGE OF WATER INTO BURNER
BURNER CONTROLS FAILURE
USE OF MORE VOLATILE SOLVENT

URR in Intermediate Level Cell

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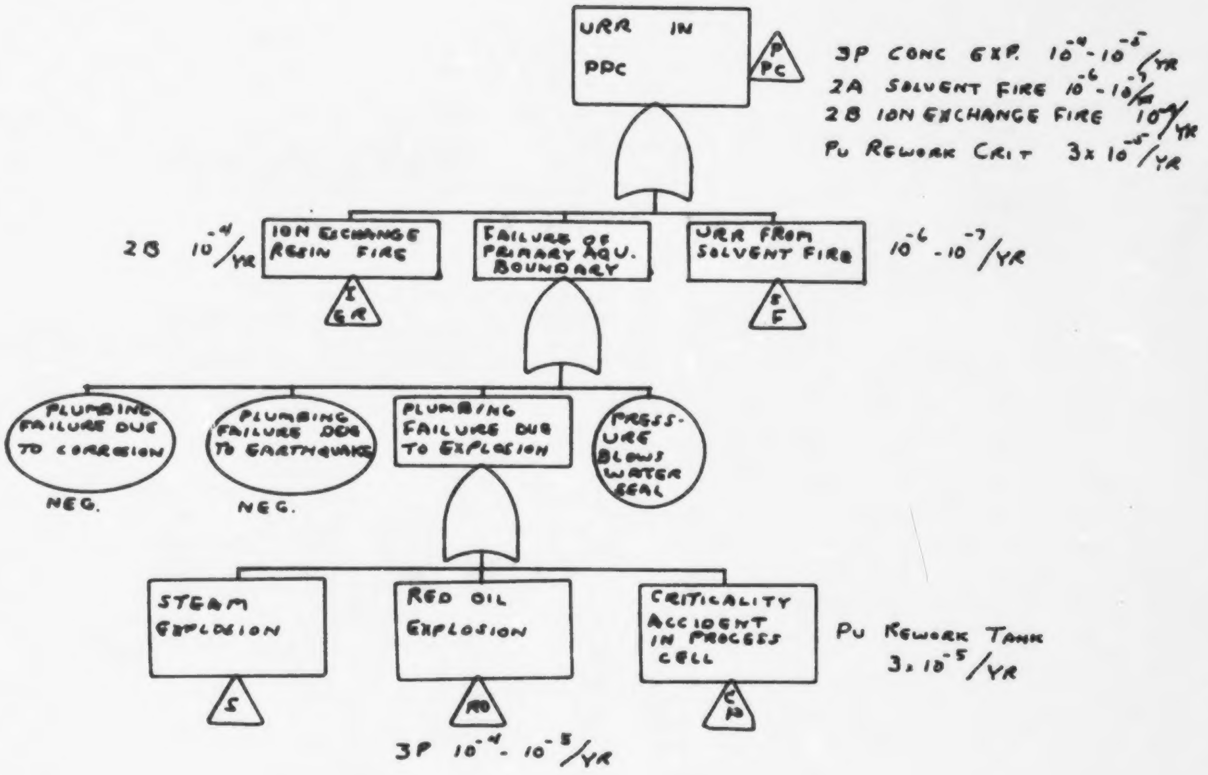
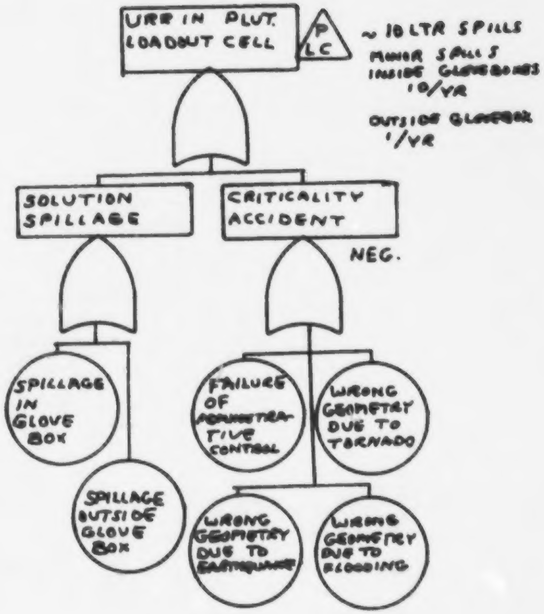
EQUIPMENT LIST

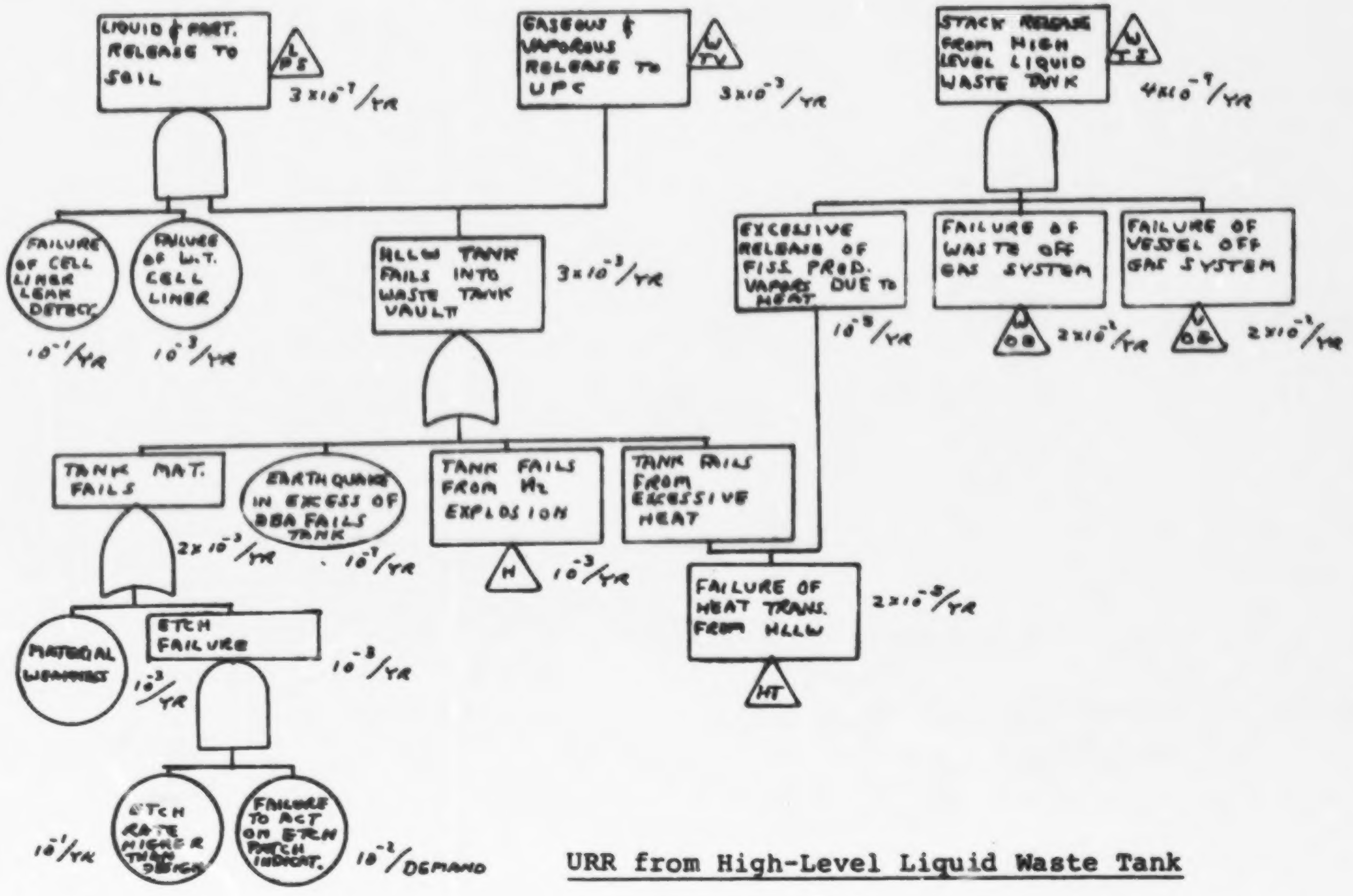

URR IN PPC
P PC
3P CONC EXP. $10^{-4}-10^{-5}$/YR
2A SOLVENT FIRE $10^{-6}-10^{-7}$/YR
2B ION EXCHANGE FIRE 10^{-4}/YR
Pu REWORK CRIT 3×10^{-5}/YR
2B 10^{-4}/YR
ION EXCHANGE RESIN FIRE
I ER
FAILURE OF PRIMARY AQU. BOUNDARY
URR FROM SOLVENT FIRE
S F
$10^{-6}-10^{-7}$/YR
PLUMBING FAILURE DUE TO CORROSION
NEG.
PLUMBING FAILURE DUE TO EARTHQUAKE
NEG.
PLUMBING FAILURE DUE TO EXPLOSION
PRESSURE BLOWS WATER SEAL
STEAM EXPLOSION
S
RED OIL EXPLOSION
RO
3P $10^{-4}-10^{-5}$/YR
CRITICALITY ACCIDENT IN PROCESS CELL
C P
Pu REWORK TANK
3×10^{-5}/YR

URR in Plutonium Process Cell

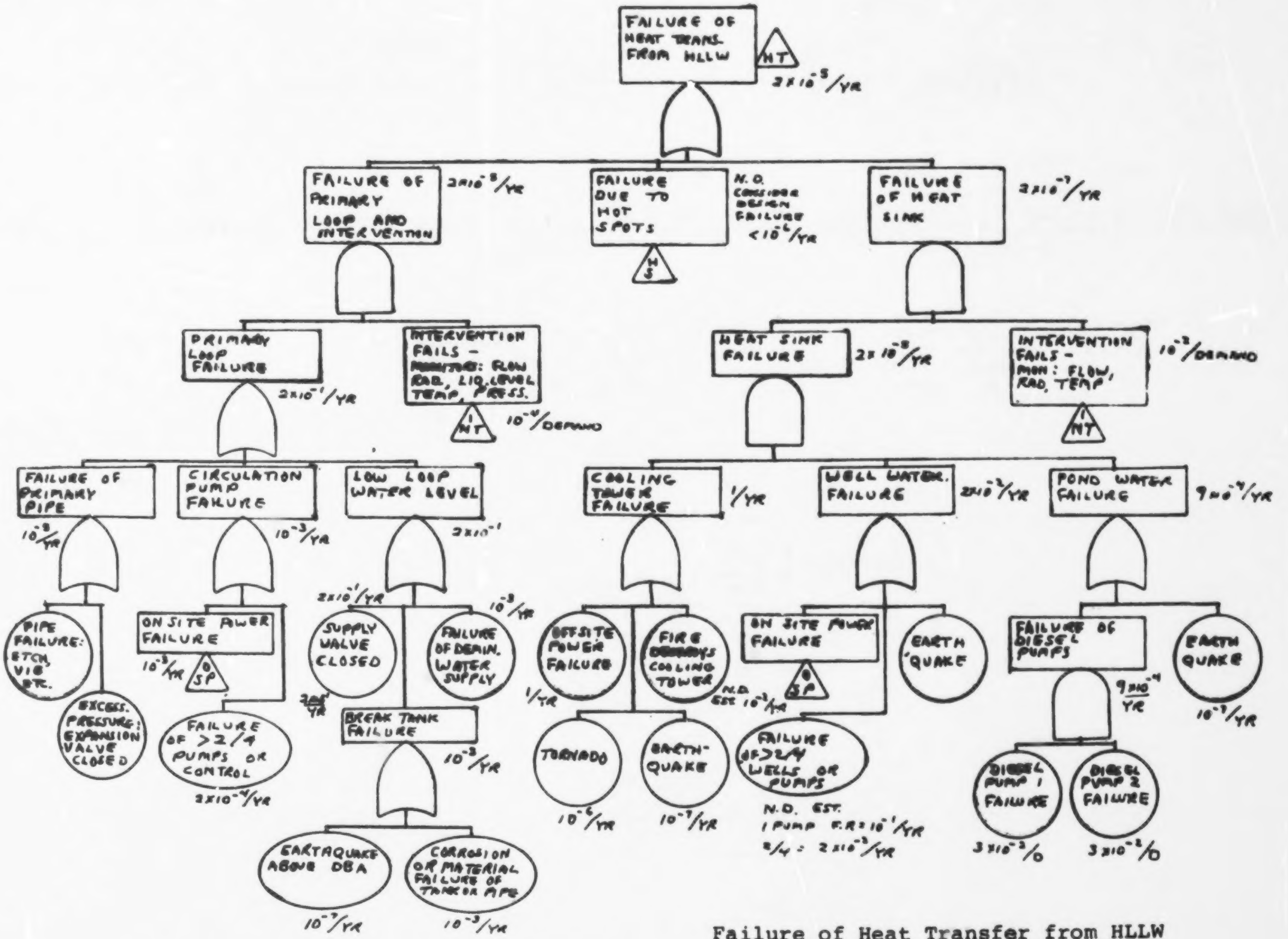
	70D 720	C W EXPANSION TANK
	70E-737	C W EXCHANGER
	70G-790A/B	C W CIRCULATION PUMPS
	PLUTONIUM	NITRATE STORAGE
	36A-301	PUMP GLOVE BOX #1
■	36A-302	PUMP GLOVE BOX #2
	36A-303	VALVE GLOVE BOX #1
■	36A-304	VALVE GLOVE BOX #2
	36A-305	SAMPLE GLOVE BOX #1
■	36A-306	SAMPLE GLOVE BOX #2
	36A-307	PN LOADOUT GLOVE BOX
	36A-308	MAINTENANCE GLOVE BOX
	36D-308A THRU F	PLUTONIUM NITRATE STORAGE TANKS
	36D-309A THRU F	PLUTONIUM NITRATE STORAGE TANKS
	36D-310 A THRU F	PLUTONIUM NITRATE STORAGE TANKS
	36D-311 A THRU F	PLUTONIUM NITRATE STORAGE TANKS
■	36D-312 A THRU F	PLUTONIUM NITRATE STORAGE TANKS
■	36D-313 A THRU F	PLUTONIUM NITRATE STORAGE TANKS
■	36D-314 A THRU F	PLUTONIUM NITRATE STORAGE TANKS
■	36D-315 A THRU F	PLUTONIUM NITRATE STORAGE TANKS
	36D-316	50 LITER NITRATE SAMPLE TANK
	36D-317	10 LTR NITRATE MEASURING TANK
	36G-308	P.N.S. TRANSFER PUMP
	36G-309	P.N.S. TRANSFER PUMP
	36G-310	P N S TRANSFER PUMP
	36G-311	P N S TRANSFER PUMP
■	36G-312	PNLO TRANSFER PUMP
■	36G-313	P N S TRANSFER PUMP
■	36G-314	P.N.S TRANSFER PUMP
■	36G-315	P N S TRANSFER PUMP
	36G-316	PNLO TRANSFER PUMP
	36G-317	SUMP PUMP PN CELL #1
■	36G-318	SUMP PUMP PN CELL #2
	75B-750A/B	TANK VAULT COOLING UNIT
■	75B-751	TANK VAULT COOLING UNIT
	75K-750	PNLO GLOVE BOX INLET FILTER
	75K-751	PNLO GLOVE BOX EXH FILTER
	75K-752	PUMP GLOVE BOX INLET FILTER
	75K-753	PUMP GLOVE BOX EXH FILTER
	75K-754	PUMP GLOVE BOX INLET FILTER
■	75K-755	PUMP GLOVE BOX INLET FILTER
■	75K-756	PUMP GLOVE BOX EXH FILTER
■	75K-757	PUMP GLOVE BOX INLET FILTER
	75K-758	VALVE GLOVE BOX INLET FILTER
	75K-759	VALVE GLOVE BOX EXH. FILTER
	75K-760	VALVE GLOVE BOX INLET FILTER
■	75K-761	VALVE GLOVE BOX INLET FILTER
■	75K-762	VALVE GLOVE BOX EXH FILTER
■	75K-763	VALVE GLOVE BOX INLET FILTER
	75K-764	SAMPLE GLOVE BOX EXH. FILTER
	75K-765	SAMPLE GLOVE BOX INLET FILTER
■	75K-766	SAMPLE GLOVE BOX EXH FILTER
■	75K-767	SAMPLE GLOVE BOX INLET FILTER
	75K-768A/B	TANK VAULT RECIRCUL. FILTER
	75K-769	EMERGENCY INLET FILTER
	80D-8105	SAMPLE VALVE POT
	80D-8106	SAMPLE VALVE POT
	80D-8107	SAMPLE VALVE POT
	80D-8108	SAMPLE VALVE POT
■	80D-8109	SAMPLE VALVE POT
■	80D-8110	SAMPLE VALVE POT
■	80D-8111	SAMPLE VALVE POT
■	80D-8112	SAMPLE VALVE POT


URR IN PLUT. LOADOUT CELL
P LC
~10 LTR SPILLS
MINOR SPILLS INSIDE GLOVEBOXES 10/YR
OUTSIDE GLOVEBOX 1/YR
SOLUTION SPILLAGE
CRITICALITY ACCIDENT
NEG.
SPILLAGE IN GLOVE BOX
SPILLAGE OUTSIDE GLOVE BOX
FAILURE OF ADMINISTRATIVE CONTROL
WRONG GEOMETRY DUE TO TORNADO
WRONG GEOMETRY DUE TO EARTHQUAKE
WRONG GEOMETRY DUE TO FLOODING

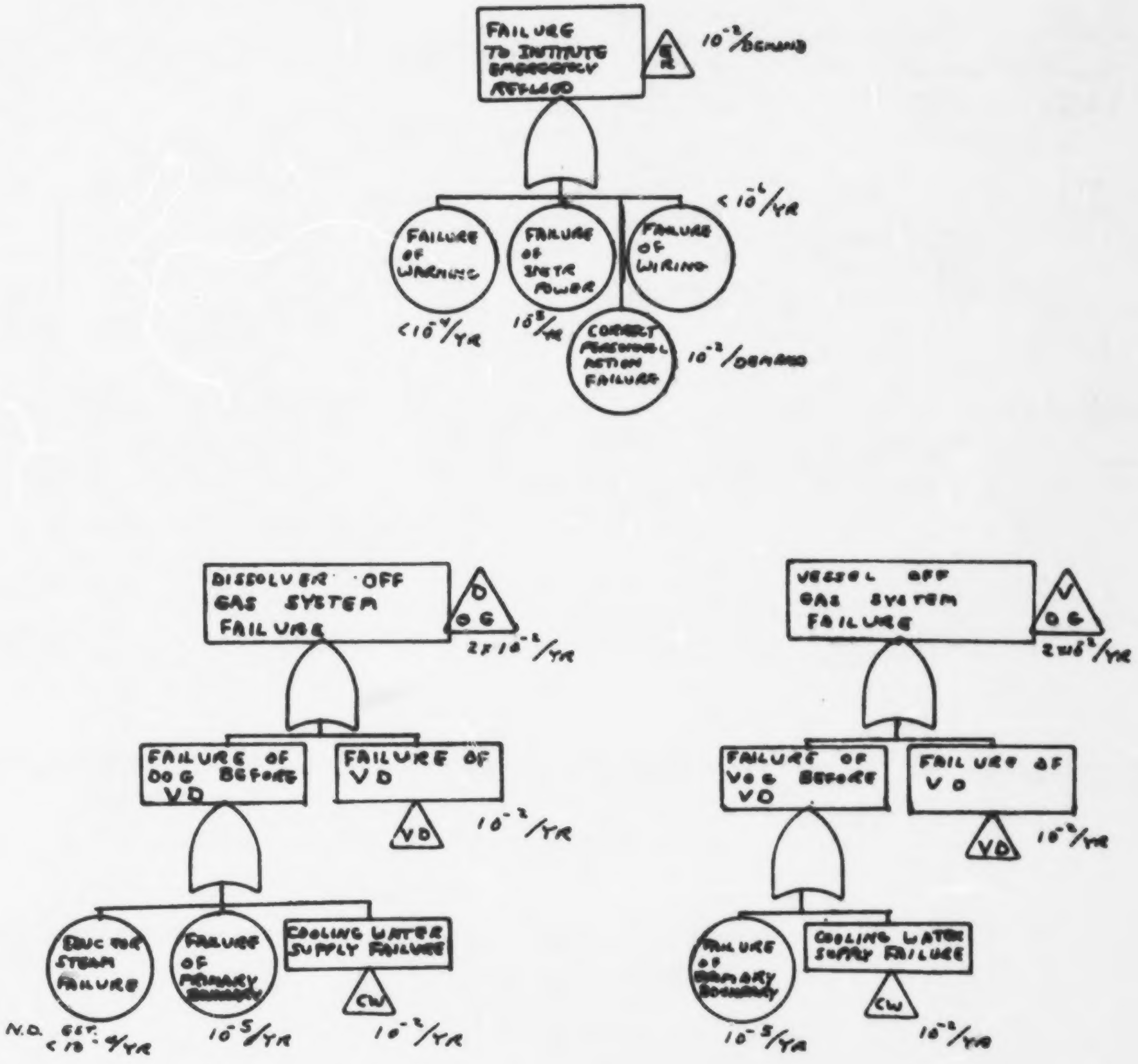
URR in Plutonium Loadout Cell


LIQUID & PART. RELEASE TO SOIL
L PS
3×10^{-7}/YR
GASEOUS & VAPOROUS RELEASE TO UPC
W TV
3×10^{-3}/YR
STACK RELEASE FROM HIGH LEVEL LIQUID WASTE TANK
W TI
4×10^{-9}/YR
FAILURE OF CELL LINER LEAK DETECT.
10^{-1}/YR
FAILURE OF W.T. CELL LINER
10^{-3}/YR
HLLW TANK FAILS INTO WASTE TANK VAULT
3×10^{-3}/YR
EXCESSIVE RELEASE OF FISS. PROD. VAPORS DUE TO HEAT
10^{-3}/YR
FAILURE OF WASTE OFF GAS SYSTEM
W OG
2×10^{-2}/YR
FAILURE OF VESSEL OFF GAS SYSTEM
V OG
2×10^{-2}/YR
TANK MAT. FAILS
2×10^{-3}/YR
EARTHQUAKE IN EXCESS OF DBA FAILS TANK
10^{-7}/YR
TANK FAILS FROM H_2 EXPLOSION
H
10^{-3}/YR
TANK FAILS FROM EXCESSIVE HEAT
FAILURE OF HEAT TRANS. FROM HLLW
HT
2×10^{-5}/YR
MATERIAL WEAKNESS
10^{-3}/YR
ETCH FAILURE
10^{-3}/YR
ETCH RATE HIGHER THAN DESIGN
10^{-1}/YR
FAILURE TO ACT ON ETCH PATCH INDICAT.
10^{-2}/DEMAND

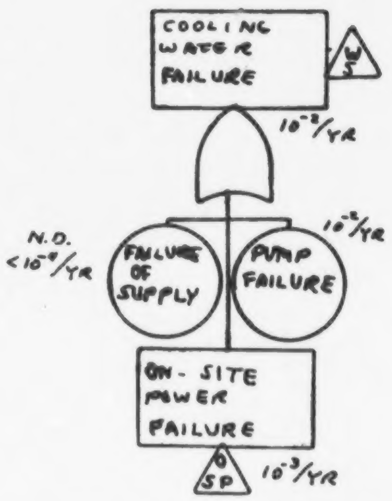
URR from High-Level Liquid Waste Tank


FAILURE OF HEAT TRANS. FROM HLLW
HT
2x10⁻⁵/YR
FAILURE OF PRIMARY LOOP AND INTERVENTION
2x10⁻⁵/YR
FAILURE DUE TO HOT SPOTS
HS
N.D. CONSIDER DESIGN FAILURE <10⁻⁶/YR
FAILURE OF HEAT SINK
2x10⁻⁷/YR
PRIMARY LOOP FAILURE
2x10⁻¹/YR
INTERVENTION FAILS - MONITORS: FLOW RAD, LIQ LEVEL TEMP, PRESS.
I NT
10⁻⁴/DEMAND
HEAT SINK FAILURE
2x10⁻⁵/YR
INTERVENTION FAILS - MON: FLOW, RAD, TEMP
I NT
10⁻²/DEMAND
FAILURE OF PRIMARY PIPE
10⁻³/YR
CIRCULATION PUMP FAILURE
10⁻³/YR
LOW LOOP WATER LEVEL
2x10⁻¹
COOLING TOWER FAILURE
1/YR
WELL WATER FAILURE
2x10⁻²/YR
POND WATER FAILURE
9x10⁻⁴/YR
PIPE FAILURE: ETCH, VIB ETC.
EXCESS. PRESSURE: EXPANSION VALVE CLOSED
ON SITE POWER FAILURE
O SP
10⁻³/YR
FAILURE OF >2/4 PUMPS OR CONTROL
2x10⁻⁴/YR
SUPPLY VALVE CLOSED
2x10⁻¹/YR
FAILURE OF DEMIN. WATER SUPPLY
10⁻³/YR
BREAK TANK FAILURE
10⁻³/YR
EARTAQUAKE ABOVE DBA
10⁻⁷/YR
CORROSION OR MATERIAL FAILURE OF TANK OR PIPE
10⁻³/YR
OFFSITE POWER FAILURE
1/YR
FIRE DESTROYS COOLING TOWER
N.D. EST 10⁻³/YR
TORNADO
10⁻⁶/YR
EARTH-QUAKE
10⁻⁷/YR
ON SITE POWER FAILURE
O SP
EARTH QUAKE
FAILURE OF >2/4 WELLS OR PUMPS
N.D. EST.
1 PUMP F.R = 10⁻¹/YR
2/4 = 2x10⁻²/YR
FAILURE OF DIESEL PUMPS
9x10⁻⁴/YR
EARTH QUAKE
10⁻⁷/YR
DIESEL PUMP 1 FAILURE
3x10⁻²/D
DIESEL PUMP 2 FAILURE
3x10⁻²/D

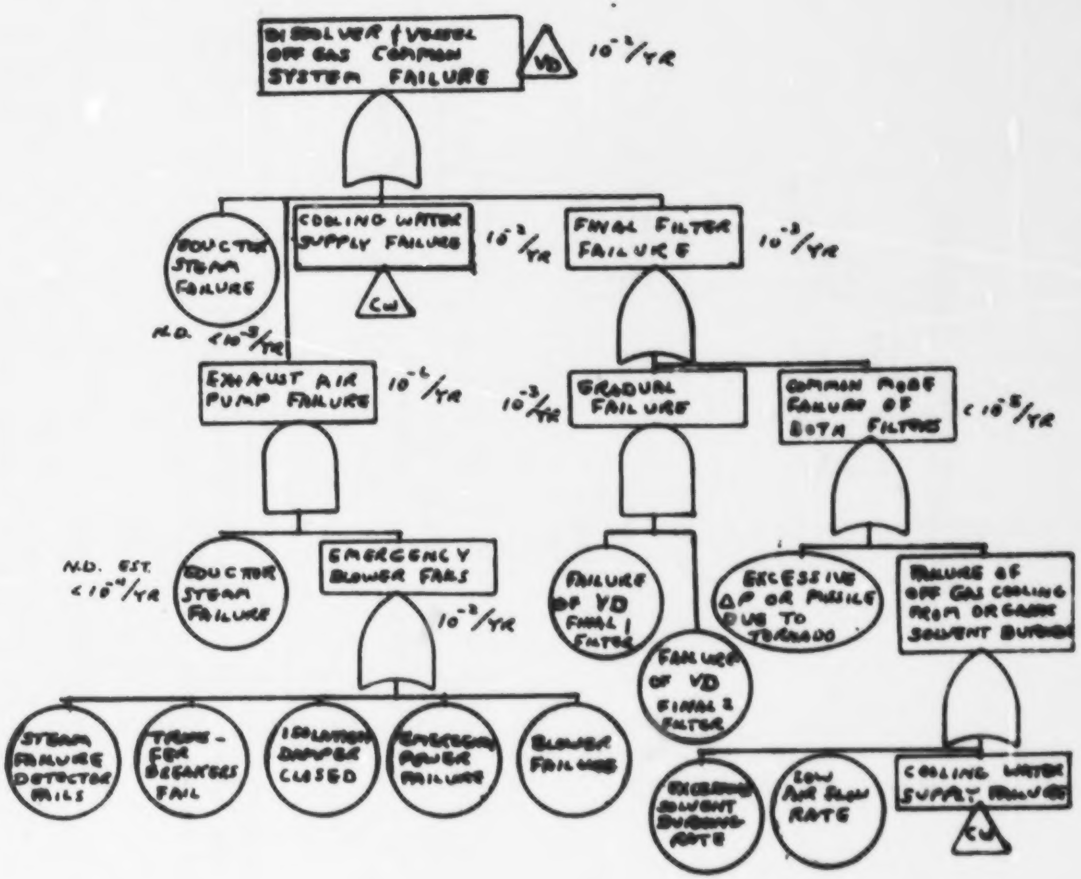
Failure of Heat Transfer from HLLW


FAILURE TO INSTITUTE EMERGENCY REFLOOD
ER
10^-2/DEMAND
FAILURE OF WARNING
<10^-4/YR
FAILURE OF SYSTR POWER
10^-3/YR
FAILURE OF WIRING
<10^-6/YR
CORRECT PERSONNEL ACTION FAILURE
10^-2/DEMAND
DISSOLVER OFF GAS SYSTEM FAILURE
DOG
2x10^-2/YR
FAILURE OF DOG BEFORE VD
FAILURE OF VD
VD
10^-2/YR
EDUCTOR STEAM FAILURE
N.D. EST. <10^-4/YR
FAILURE OF PRIMARY BOUNDARY
10^-5/YR
COOLING WATER SUPPLY FAILURE
CW
10^-2/YR
VESSEL OFF GAS SYSTEM FAILURE
VOG
2x10^-2/YR
FAILURE OF VOG BEFORE VD
FAILURE OF VD
VD
10^-2/YR
FAILURE OF PRIMARY BOUNDARY
10^-5/YR
COOLING WATER SUPPLY FAILURE
CW
10^-2/YR

Failure to Institute Emergency Reflood

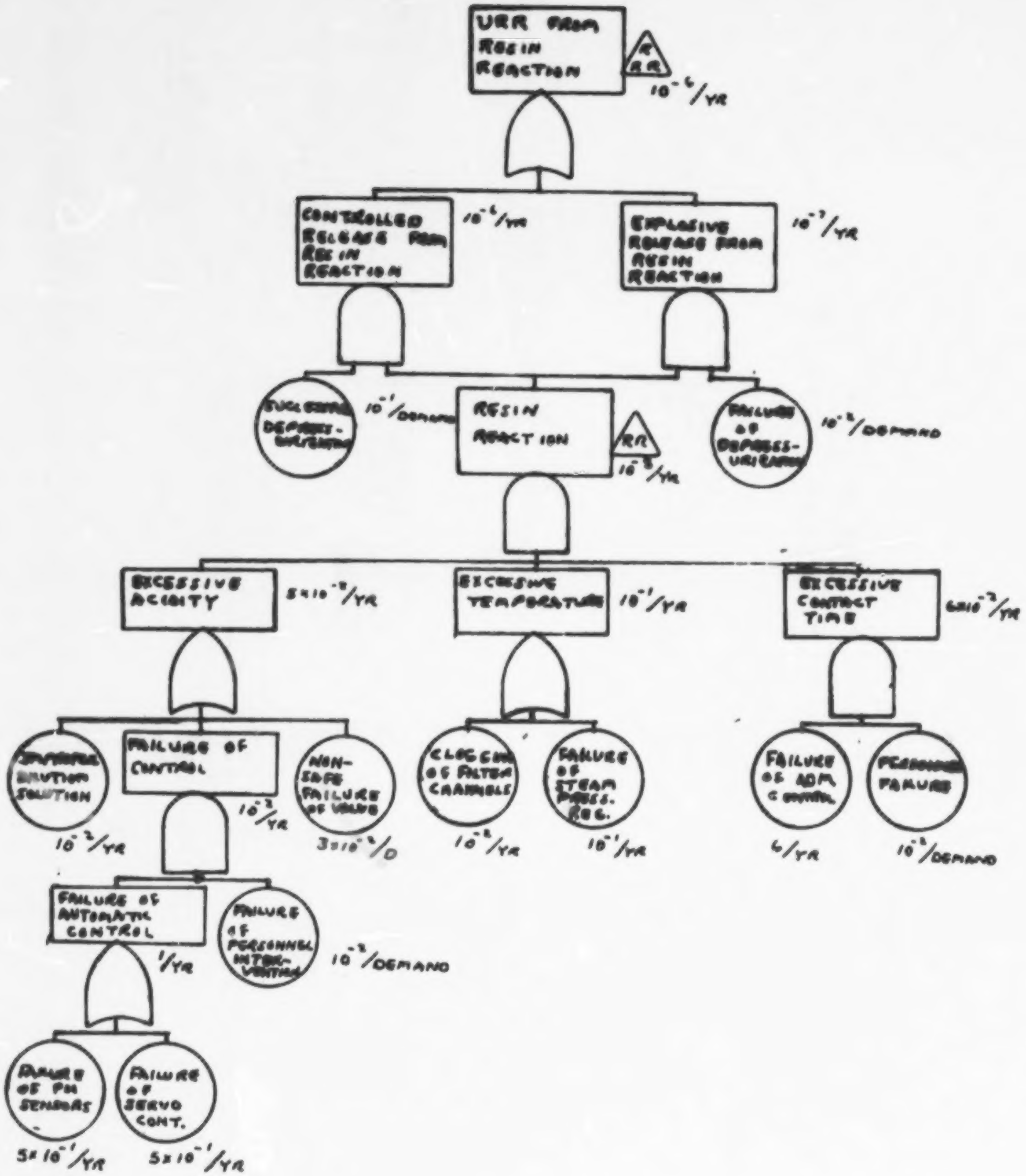

COOLING WATER FAILURE
WS
10^{-2}/YR
N.D. $<10^{-4}$/YR
FAILURE OF SUPPLY
PUMP FAILURE
10^{-2}/YR
ON-SITE POWER FAILURE
SP
10^{-3}/YR

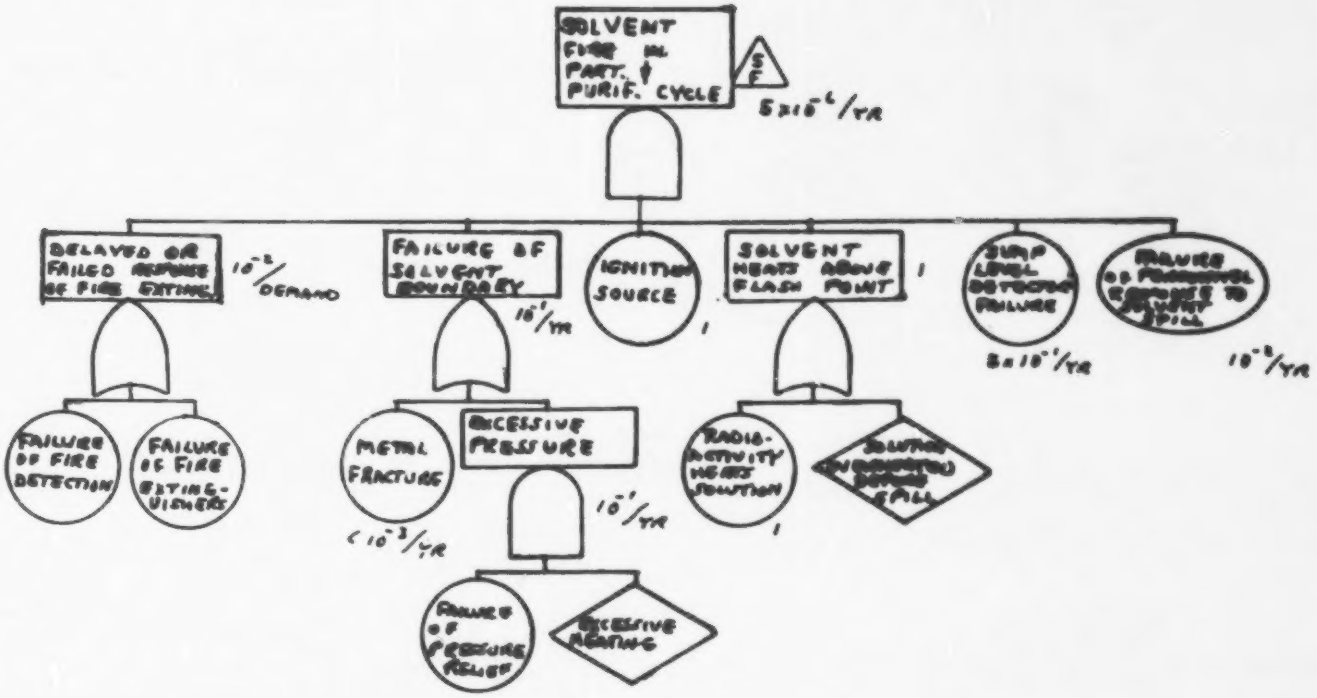
Cooling Water Failure


DISSOLVER & VESSEL OFF GAS COMMON SYSTEM FAILURE
VD
10^{-2}/YR
EDUCTOR STEAM FAILURE
N.D. $<10^{-5}$/YR
COOLING WATER SUPPLY FAILURE
CW
10^{-2}/YR
FINAL FILTER FAILURE
10^{-3}/YR
EXHAUST AIR PUMP FAILURE
10^{-6}/YR
GRADUAL FAILURE
10^{-3}/YR
COMMON MODE FAILURE OF BOTH FILTERS
$<10^{-5}$/YR
N.D. EST. $<10^{-4}$/YR
EDUCTOR STEAM FAILURE
EMERGENCY BLOWER FAILS
10^{-2}/YR
FAILURE OF VD FINAL 1 FILTER
FAILURE OF VD FINAL 2 FILTER
EXCESSIVE ΔP OR MISSILE DUE TO TORNADO
FAILURE OF OFF GAS COOLING FROM ORGANIC SOLVENT BURNING
STEAM FAILURE DETECTOR FAILS
TRANSFER BREAKERS FAIL
ISOLATION DAMPER CLOSED
EMERGENCY POWER FAILURE
BLOWER FAILURE
EXCESSIVE SOLVENT BURNING RATE
LOW AIR FLOW RATE
COOLING WATER SUPPLY FAILURE
CW

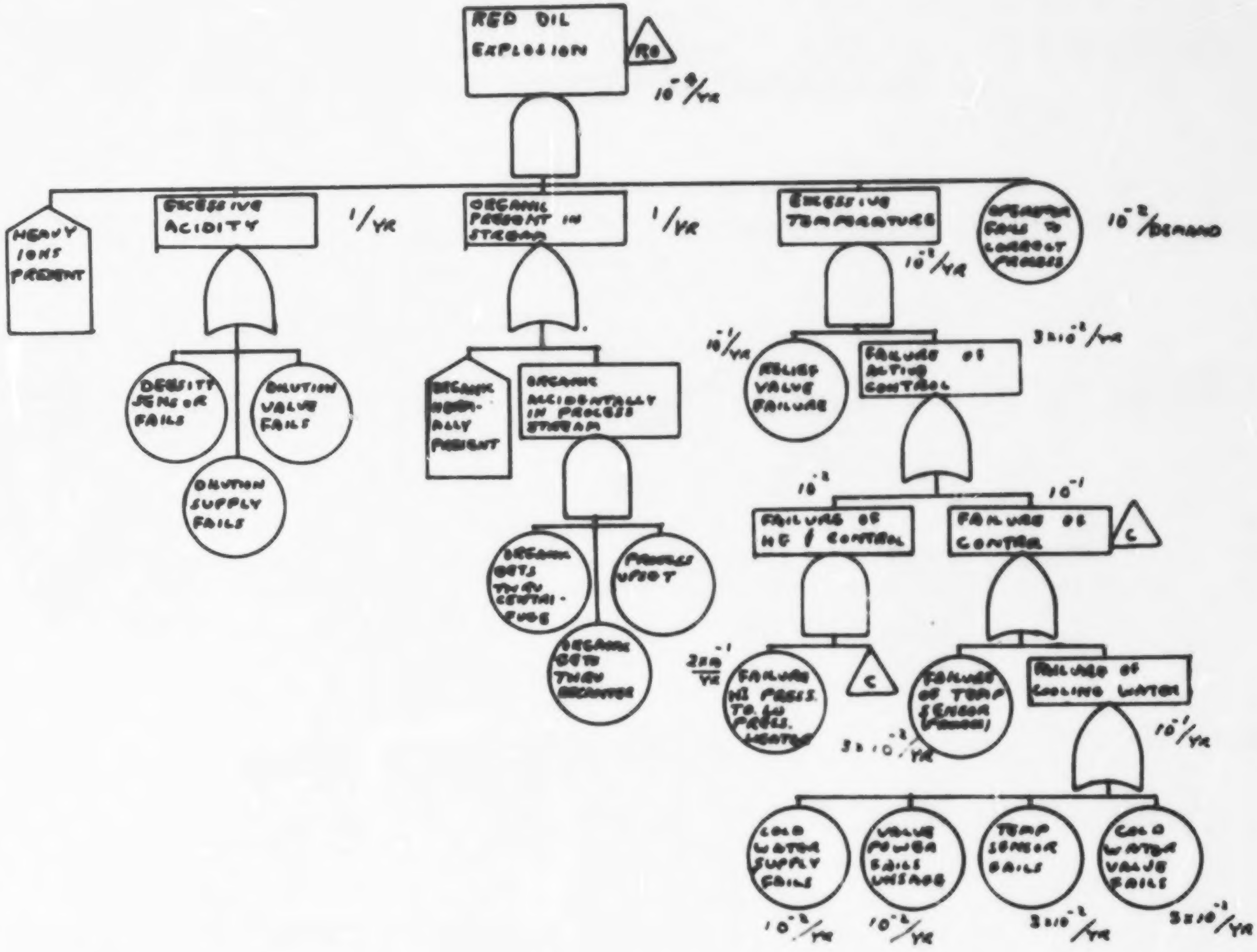
Dissolver and Vessel Off-Gas Common System

URR from Resin Reaction

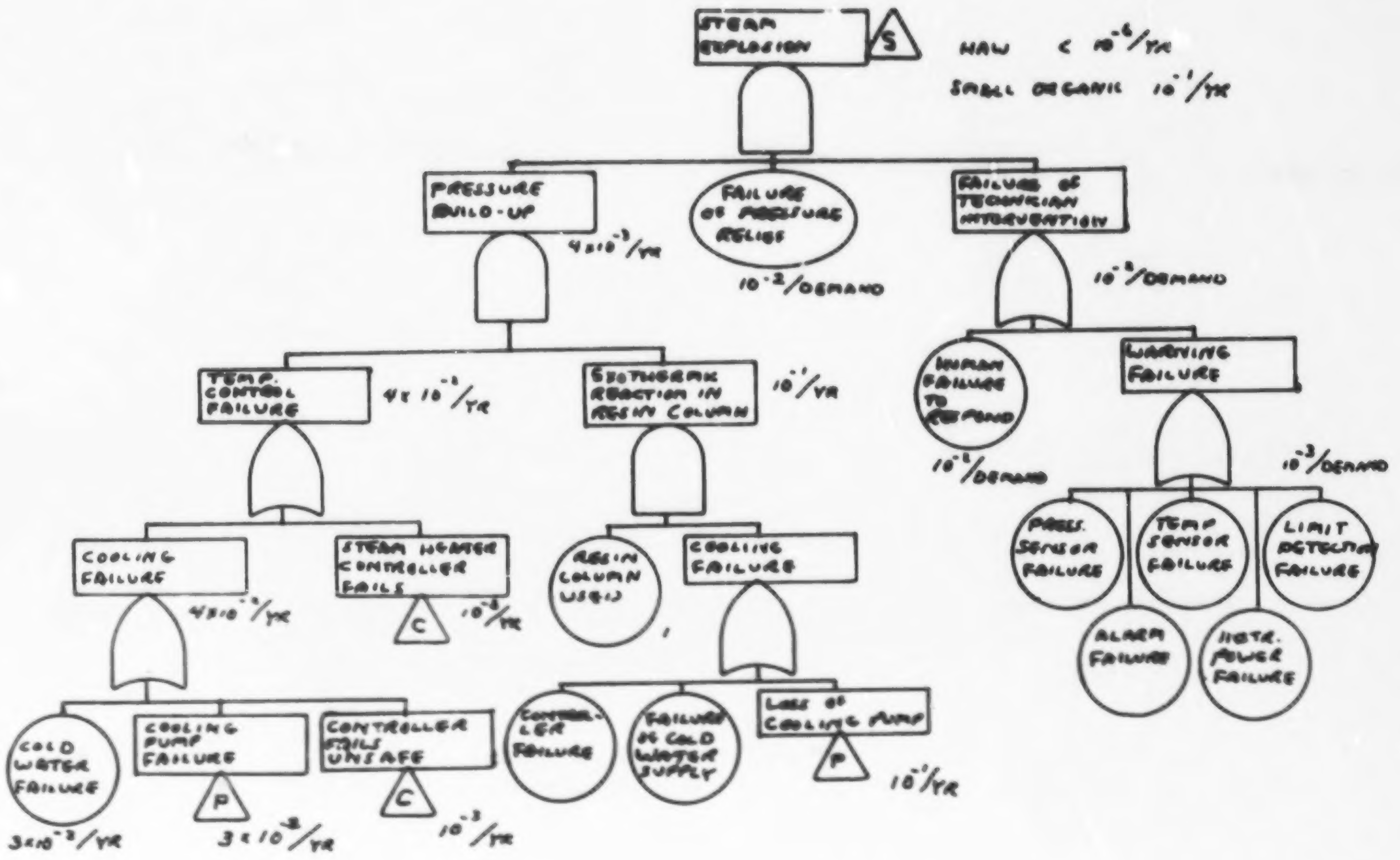

URR FROM RESIN REACTION
R RR
10^{-6}/YR
CONTROLLED RELEASE FROM RESIN REACTION
10^{-6}/YR
EXPLOSIVE RELEASE FROM RESIN REACTION
10^{-7}/YR
SUCCESSFUL DEPRESSURIZATION
10^{-1}/DEMAND
RESIN REACTION
RR
10^{-5}/YR
FAILURE OF DEPRESSURIZATION
10^{-2}/DEMAND
EXCESSIVE ACIDITY
5×10^{-3}/YR
EXCESSIVE TEMPERATURE
10^{-1}/YR
EXCESSIVE CONTACT TIME
6×10^{-2}/YR
[illegible] ELUTION SOLUTION
10^{-2}/YR
FAILURE OF CONTROL
10^{-2}/YR
NON-SAFE FAILURE OF VALVE
3×10^{-2}/D
CLOGGING OF FILTER CHANNELS
10^{-2}/YR
FAILURE OF STEAM PRESS. REG.
10^{-1}/YR
FAILURE OF ADM CONTROL
6/YR
PERSONNEL FAILURE
10^{-2}/DEMAND
FAILURE OF AUTOMATIC CONTROL
1/YR
FAILURE OF PERSONNEL INTER-VENTION
10^{-2}/DEMAND
FAILURE OF PH SENSORS
5×10^{-1}/YR
FAILURE OF SERVO CONT.
5×10^{-1}/YR


SOLVENT FIRE IN PART. & PURIF. CYCLE
S F
5×10^{-6}/YR
DELAYED OR FAILED RESPONSE OF FIRE EXTING.
10^{-2}/DEMAND
FAILURE OF FIRE DETECTION
FAILURE OF FIRE EXTINGUISHERS
FAILURE OF SOLVENT BOUNDARY
10^{-1}/YR
METAL FRACTURE
$< 10^{-3}$/YR
EXCESSIVE PRESSURE
10^{-1}/YR
FAILURE OF PRESSURE RELIEF
EXCESSIVE HEATING
IGNITION SOURCE
1
SOLVENT HEATS ABOVE FLASH POINT
1
RADIOACTIVITY HEATS SOLUTION
1
SOLUTION
SPILL
SUMP LEVEL DETECTOR FAILURE
3×10^{-1}/YR
FAILURE OF PERSONNEL RESPONSE TO SOLVENT SPILL
10^{-2}/YR

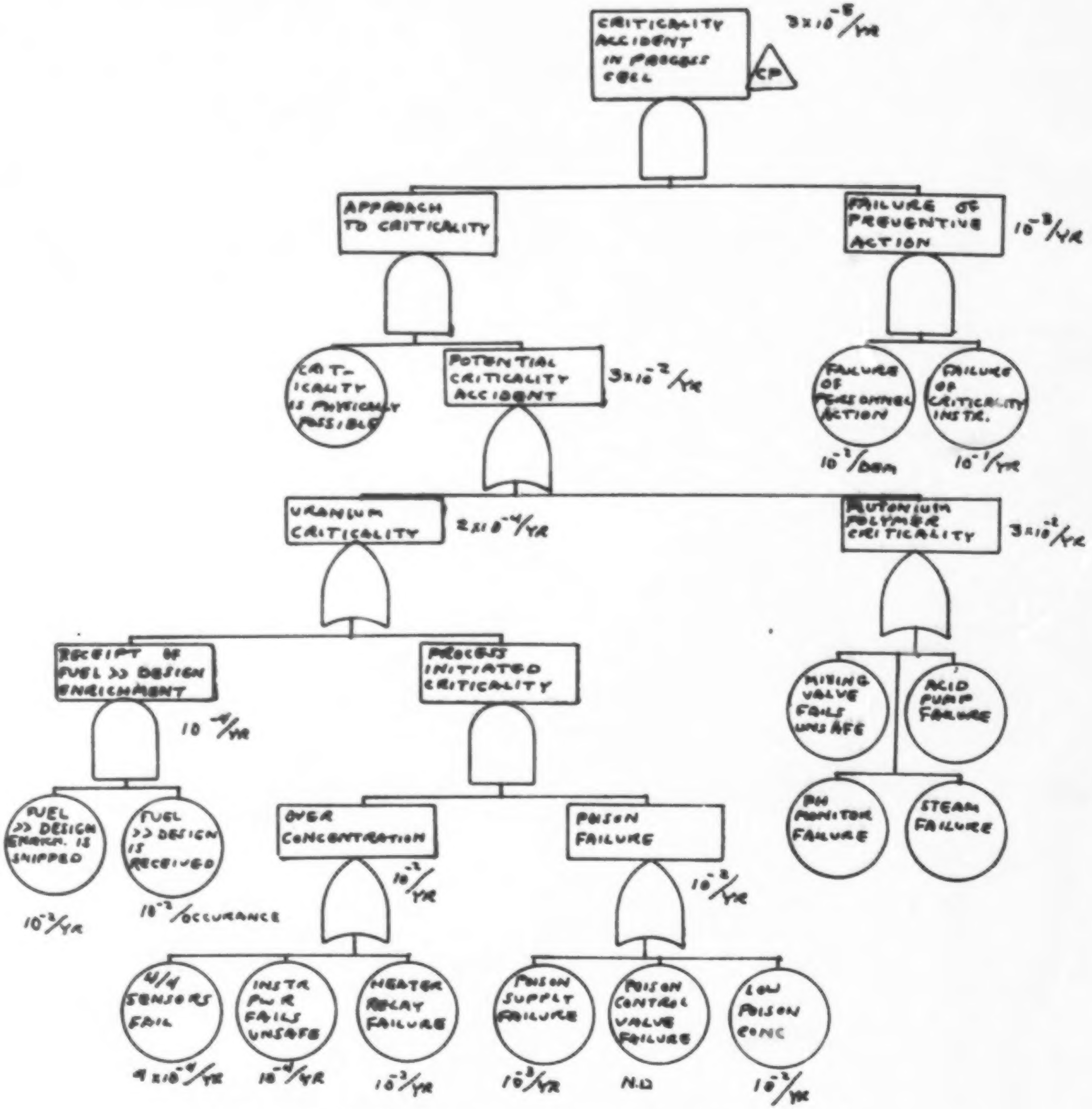
Solvent Fire in Partition and Purification


RED OIL EXPLOSION
RO
10^-9/YR
HEAVY IONS PRESENT
EXCESSIVE ACIDITY
1/YR
DENSITY SENSOR FAILS
DILUTION VALVE FAILS
DILUTION SUPPLY FAILS
ORGANIC PRESENT IN STREAM
1/YR
ORGANIC NORMALLY PRESENT
ORGANIC ACCIDENTALLY IN PROCESS STREAM
ORGANIC GETS THRU CENTRIFUGE
PROCESS UPSET
ORGANIC GETS THRU DECANTER
EXCESSIVE TEMPERATURE
10^-3/YR
OPERATOR FAILS TO CORRECT PROCESS
10^-2/DEMAND
10^-1/YR
RELIEF VALVE FAILURE
FAILURE OF ACTING CONTROL
3x10^-2/YR
10^-2
FAILURE OF HP / CONTROL
10^-1
FAILURE OF CONTROL
C
2x10^-1/YR
FAILURE HI PRESS. TO LO PRESS. HEATER
C
FAILURE OF TEMP SENSOR
3x10^-2/YR
FAILURE OF COOLING WATER
10^-1/YR
COLD WATER SUPPLY FAILS
10^-2/YR
VALVE POWER FAILS UNSAFE
10^-2/YR
TEMP SENSOR FAILS
3x10^-2/YR
COLD WATER VALVE FAILS
3x10^-2/YR

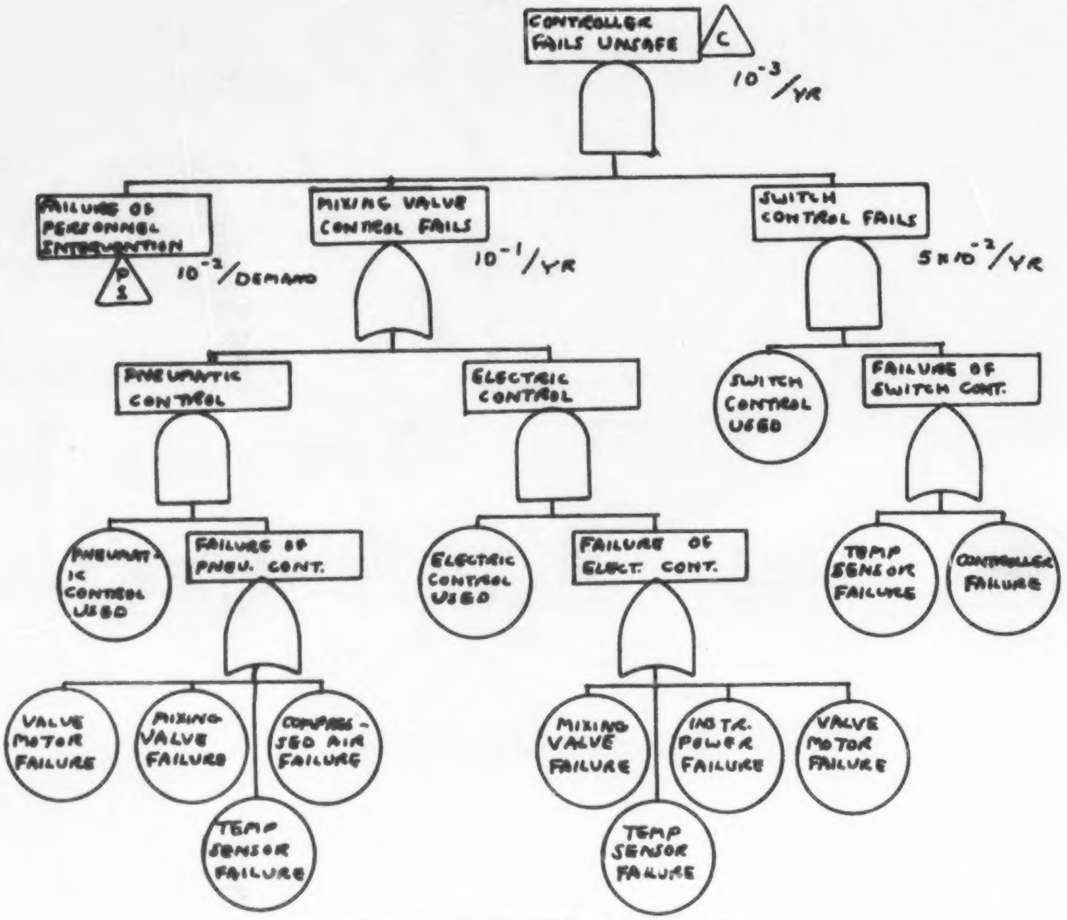
Red Oil Explosion


STEAM EXPLOSION
S
HAW < 10^-6/YR
SMALL ORGANIC 10^-1/YR
PRESSURE BUILD-UP
4x10^-3/YR
FAILURE OF PRESSURE RELIEF
10^-2/DEMAND
FAILURE OF TECHNICIAN INTERVENTION
10^-3/DEMAND
TEMP. CONTROL FAILURE
4x 10^-3/YR
EXOTHERMIC REACTION IN RESIN COLUMN
10^-1/YR
HUMAN FAILURE TO RESPOND
10^-2/DEMAND
WARNING FAILURE
10^-3/DEMAND
COOLING FAILURE
4x10^-2/YR
STEAM HEATER CONTROLLER FAILS
C
10^-3/YR
RESIN COLUMN USED
COOLING FAILURE
PRESS. SENSOR FAILURE
TEMP SENSOR FAILURE
LIMIT DETECTION FAILURE
ALARM FAILURE
INSTR. POWER FAILURE
COLD WATER FAILURE
3x10^-2/YR
COOLING PUMP FAILURE
P
3x10^-2/YR
CONTROLLER FAILS UNSAFE
C
10^-3/YR
CONTROLLER FAILURE
FAILURE OF COLD WATER SUPPLY
LOSS OF COOLING PUMP
P
10^-1/YR

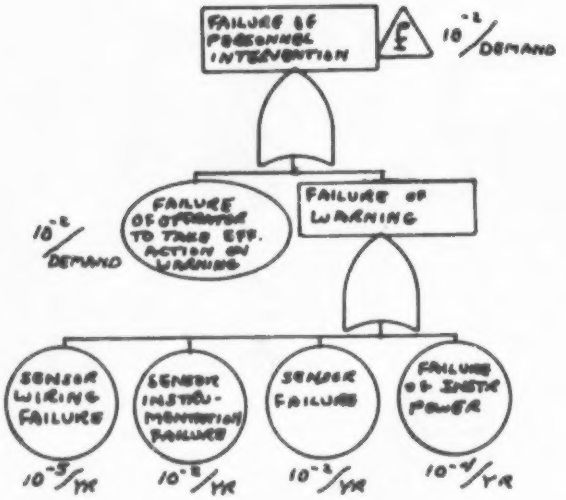
Steam Explosion


CRITICALITY ACCIDENT IN PROCESS CELL
CP
3x10^-5/YR
APPROACH TO CRITICALITY
FAILURE OF PREVENTIVE ACTION
10^-3/YR
CRIT-ICALITY IS PHYSICALLY POSSIBLE
POTENTIAL CRITICALITY ACCIDENT
3x10^-2/YR
FAILURE OF PERSONNEL ACTION
10^-2/DEM
FAILURE OF CRITICALITY INSTR.
10^-1/YR
URANIUM CRITICALITY
2x10^-4/YR
PLUTONIUM POLYMER CRITICALITY
3x10^-2/YR
RECEIPT OF FUEL >> DESIGN ENRICHMENT
10^-4/YR
PROCESS INITIATED CRITICALITY
MIXING VALVE FAILS UNSAFE
ACID PUMP FAILURE
PH MONITOR FAILURE
STEAM FAILURE
FUEL >> DESIGN ENRICH. IS SHIPPED
10^-2/YR
FUEL >> DESIGN IS RECEIVED
10^-2/OCCURANCE
OVER CONCENTRATION
10^-2/YR
POISON FAILURE
10^-3/YR
4/4 SENSORS FAIL
4x10^-4/YR
INSTR PWR FAILS UNSAFE
10^-4/YR
HEATER RELAY FAILURE
10^-3/YR
POISON SUPPLY FAILURE
10^-3/YR
POISON CONTROL VALVE FAILURE
N.D
LOW POISON CONC
10^-2/YR

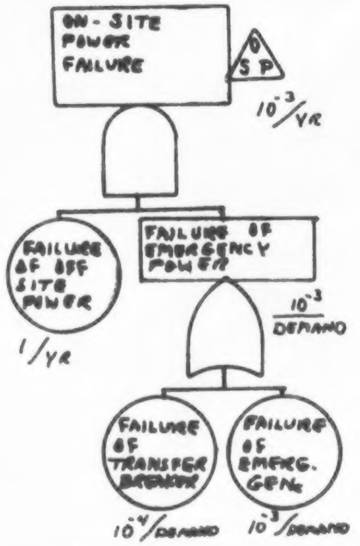
Criticality Accident in Process Cell


CONTROLLER FAILS UNSAFE
C
10^{-3}/YR
FAILURE OF PERSONNEL INTERVENTION
P I
10^{-2}/DEMAND
MIXING VALVE CONTROL FAILS
10^{-1}/YR
SWITCH CONTROL FAILS
5×10^{-2}/YR
PNEUMATIC CONTROL
ELECTRIC CONTROL
SWITCH CONTROL USED
FAILURE OF SWITCH CONT.
PNEUMATIC CONTROL USED
FAILURE OF PNEU. CONT.
ELECTRIC CONTROL USED
FAILURE OF ELECT. CONT.
TEMP SENSOR FAILURE
CONTROLLER FAILURE
VALVE MOTOR FAILURE
MIXING VALVE FAILURE
COMPRESSED AIR FAILURE
TEMP SENSOR FAILURE
MIXING VALVE FAILURE
INSTR. POWER FAILURE
VALVE MOTOR FAILURE
TEMP SENSOR FAILURE

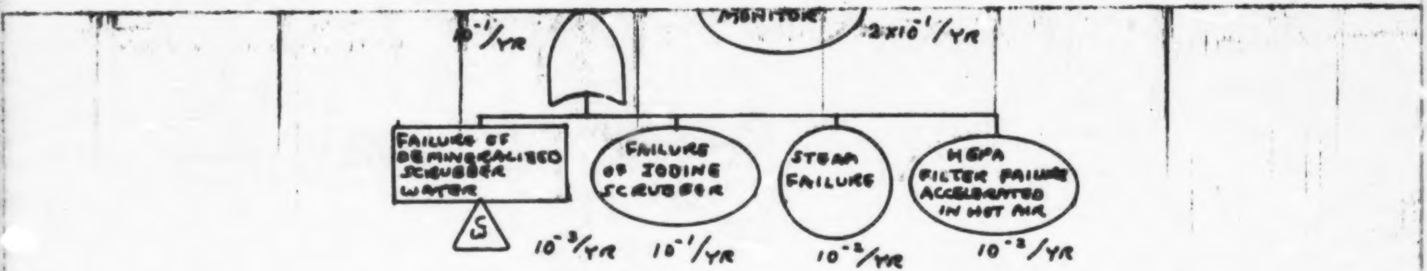
Controller Fails Unsafe


FAILURE OF PERSONNEL INTERVENTION
P I
10^{-2}/DEMAND
FAILURE OF OPERATOR TO TAKE EFF. ACTION ON WARNING
10^{-2}/DEMAND
FAILURE OF WARNING
SENSOR WIRING FAILURE
SENSOR INSTRUMENTATION FAILURE
SENSOR FAILURE
FAILURE OF INSTR POWER
10^{-5}/YR
10^{-3}/YR
10^{-2}/YR
10^{-4}/YR

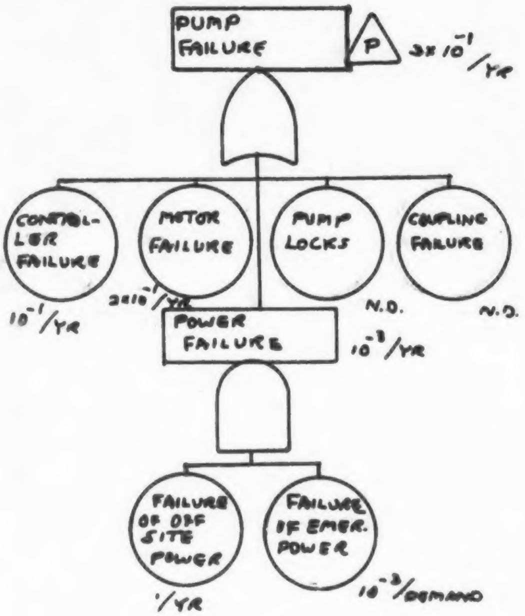
Failure of Personnel Intervention


ON-SITE POWER FAILURE
O S P
10^{-3}/YR
FAILURE OF OFF SITE POWER
1/YR
FAILURE OF EMERGENCY POWER
10^{-3}/DEMAND
FAILURE OF TRANSFER BREAKER
10^{-4}/DEMAND
FAILURE OF EMERG. GEN.
10^{-3}/DEMAND

On-Site Power Failure


MONITOR
10⁻¹/YR
2x10⁻¹/YR
FAILURE OF DEMINERALIZED SCRUBBER WATER
S
FAILURE OF IODINE SCRUBBER
STEAM FAILURE
HEPA FILTER FAILURE ACCELERATED IN HOT AIR
10⁻³/YR
10⁻¹/YR
10⁻²/YR
10⁻²/YR

Acid Fraction Overhead


PUMP FAILURE
P
3x10⁻¹/YR
CONTROLLER FAILURE
MOTOR FAILURE
PUMP LOCKS
COUPLING FAILURE
10⁻¹/YR
2x10⁻¹/YR
N.D.
N.D.
POWER FAILURE
10⁻³/YR
FAILURE OF OFF SITE POWER
FAILURE OF EMER. POWER
1/YR
10⁻³/DEMAND

Pump Failure

APPENDIX C

Descriptions of Accidents Experienced in the Nuclear Energy Field and Chemical Industry Relating to Anticipated Credible Events at a Fuel Reprocessing Plant.

This Appendix reviews the types of accidents which have occurred in AEC operational activities relating to operations performed at a nuclear fuels reprocessing plant. The information is drawn from AEC field office reports as described in USAEC Report WASH-1192[34] and supplemental information supplied by the USAEC Division of Operational Safety.[54] There is a tendency to emphasize exposures and contamination from criticality incidents in discussions concerning radiation hazards because of the AEC's involvement. These occurrences, however, are in the minority in the overall picture of potential environmental impact. By far, the majority of potential incidents which could occur during activities at a fuel reprocessing plant, that might have an environmental impact are comparable, in kind, to those occurring in the chemical industry. To broaden the statistical base for accident probability estimates for this study, available relevant accident case histories from the chemical industry are also included in this Appendix.[21] These case histories, covering the period 1951 through 1972 are voluntarily submitted to the Manufacturing Chemists Association for publication in an endeavor to improve plant safety in the industry. This compilation, to be sure, is not complete. It does, however, indicate the types of accidents that have occurred in the operation of a reprocessing plant or related facility and was utilized in selecting the hypothetical accidents considered in this study.

The abbreviations used for the USAEC field offices are as follows:

AL	Albuquerque Operations Office
BH	Brookhaven Office

CH	Chicago Operations Office
GJ	Grand Junction Office
HA	Hanford Operations Office
ID	Idaho Operations Office
LAR	Lockland Aircraft Reactors Office
NV	Nevada Operations Office
NY	New York Operations Office
OR	Oak Ridge Operations Office
PNR	Pittsburgh Naval Reactors Office
RL	Richland Operations Office (Formerly HA)
SAN	San Francisco Operations Office
SNR	Schenectady Naval Reactors Office
SR	Savannah River Operations Office
SNPO-C	Space Nuclear Propulsion Office-Cleveland
SNPO-N	Space Nuclear Propulsion Office-Nevada

1. Criticality Accidents

a. Oak Ridge, Tenn., June 16, 1958

A nuclear accident occurred in a 55-gallon stainless steel drum in a processing area in which enriched uranium is recovered from various materials by chemical methods in a complex of equipment. This recovery process was being remodeled at the time of the accident.

The incident occurred while they were draining material thought to be water from safe 5-inch storage pipes into an unsafe drum.

Eight employees were in the vicinity of the drum carrying out routine plant operations and maintenance. A chemical operator was participating in the leak testing which inadvertently set off the reaction. He was within three to six feet of the drum, while the other seven employees were from 15 to 50 feet away.

Using special post hoc methods for determining the neutron and gamma exposures of the employees involved, it was estimated that the eight men received: 461 rem, 428 rem, 413 rem, 341 rem, 298 rem, 86 rem, 86 rem, and 29 rem.

Area contamination was slight, with decontamination costs amounting to less than $1,000.

During this incident 1.3×10^{18} fissions occurred.

b. Los Alamos, N. Mex., Dec. 30, 1958

The chemical operator introduced what was believed to be a dilute plutonium solution from one tank into another known to contain more plutonium in emulsion. Solids containing plutonium were probably washed from the bottom of the first tank with nitric acid and the resultant mixture of nitric acid and plutonium-bearing solids was added to the tank containing the emulsion. A criticality excursion occurred immediately after starting the motor to a propeller type stirrer at the bottom of the second tank.

The operator fell from the low stepladder on which he was standing and stumbled out of the door into the snow. A second chemical operator in an adjoining room had seen a flash, which probably resulted from a short circuit when the motor to the stirrer started, and went to the man's assistance. The accident victim mumbled he felt as though he was burning up. Because of this, it was assumed that there had been a chemical accident with a probable acid or plutonium exposure. There was no realization that a criticality accident had occurred for a number of minutes. The quantity of plutonium which actually was present in the tank was about ten times more than was supposed to be there at any time during the procedure.

The employee died 35 hours later from the effects of a radiation exposure with the whole-body dose calculated to be 12,000 rem ±.

Two other employees received radiation exposure of 134 and 53 rem, respectively. Property damage was negligible. (See TID-5360, Suppl. 2, p. 30; USAEC Serious Accidents Issue #143, 1-22-59.)

c. Idaho Falls, Idaho, Oct. 16, 1959

A nuclear incident occurred in a process equipment waste collection tank when an accidental transfer was made of about 200 liters of uranyl nitrate solution, containing about 34 kilograms of enriched uranium (91 percent U^{235}), from critically safe process storage tanks to a geometrically unsafe tank through a line formerly used for waste transfers.

Limited visual inspections and test indicated that no significant property damage or loss resulted beyond the approximately $60,000 cost to recover contaminated uranium solution resulting from the incident.

Of the 21 personnel directly involved in this incident, seven received external exposures to radiation. The exposures were 8, 6, 3.95, 1.50, 1.38, 1.17, and 1.17 rem. Two individuals also received external exposures to the skin of 50 rem and 32 rem. No medical treatment was required for the 21 personnel involved.

d. Idaho Falls, Idaho, Jan. 25, 1961

A nuclear excursion of approximately 6×10^{17} fissions occurred in a first-cycle product evaporator at a chemical processing plant. The criticality accident resulted when a solution of enriched uranyl nitrate accidentally surged from a geometrically safe section of the evaporator into the upper critically unsafe, vapor disengagement section. The accident occurred behind thick concrete walls in a processing cell which is part of the first cycle for processing highly radioactive spent-fuel elements.

Personnel response to the radiation alarms and the evacuation signal was prompt and orderly.

Analyses of badges from 65 individuals indicated a maximum exposure of 55 millirem gamma and 0 beta. The maximum thermal neutron exposure detected in the badges analyzed was less than 10 millirem. Analyses of nuclear accident dosimeters indicated that there was negligible fast neutron flux associated with personnel exposures.

The radioactivity released to the atmosphere as a result of the accident was about twice normal background when it left the area. Loss of $6,000 resulted from cleanup of the incident.

e. Richland, Wash., Apr. 7, 1962

An unplanned nuclear excursion occurred in a plutonium processing facility because of the inadvertent accumulation of approximately 1500 grams of plutonium in 45-50 liters of dilute nitric acid solution in a 69-liter glass transfer tank. The sequence of events which led to the accumulation of the plutonium in the tank cannot be stated positively. However, it is believed that, when a tank valve was opened, the solution from another process vessel overflowed to a sump and was drawn into the transfer tank through a temporary line between this tank and the sump.

When the excursion occurred, radiation and evacuation alarms sounded. All but three employees left the building immediately, according to well-prepared and well-rehearsed evacuation plans. Fortunately, they were not in close proximity to the involved system nor in a high radiation field.

The course of the nuclear reaction involved initial criticality (10^{16} fissions); a subsidence; one or more later peaks; and after approximately one-half hour, a declining rate of fission, which terminated in a subcritical condition 37 hours later. The total number of fissions was approximately 8×10^{17}.

Of the 22 persons in the building at the time, only four employees, those who were in the room with the system, were hospitalized for observation. Three of them were the system operators, who were in close proximity to the excursion, and who received estimated radiation doses of 110, 43, and 19 rem. None of them showed symptoms definitely referable to their radiation exposures. The fourth was sent to the hospital only because he was in the room at the time of the incident.

Some fission product activity, airborne via the vent system and the exhaust stack, was detected in the atmosphere for a brief period after the accident. The physical damage amounted to less than $1,000.

f. Wood River Junction, R.I., July 24, 1964

Because of startup difficulties an unusually large amount of highly enriched uranium-contaminated trichloroethane (TCE) had accumulated at United Nuclear's Scrap Recovery Plant. The recovery was by mixing the TCE with a sodium carbonate solution. On the day of the incident the process was shifted to an 18 in. dia. by 25 in. deep tank to try to catch up with the backlog. The plant evaporator failed resulting in a plug of uranium nitrate crystals in the converting line. This plug was dissolved with steam and the concentrated solution was drained into a polyethylene bottle. This bottle was mistaken for trichlorethane and the operator poured it into the tank of solution. Criticality was reached in a burst of 10^{17} fissions creating a flash of light. 1/5 of the solution was ejected and the operator knocked to the floor. He ran to the emergency building 200 yards away but having received about 10,000 rad, died 49 hours later.

Later two men entered the area to drain the solution into safe containers. When the stirrer was turned off, the geometry change resulted in a second criticality of 2-3x 10^{16} fissions and these men received 60 and 100 rads.

g. Oak Ridge, Tenn., Jan. 30, 1968

Unexpected criticality was achieved in a volume of an aqueous solution of a salt of U^{233} during a series of routine critical experiments in progress in a well-shielded assembly area of a critical experiments facility. The criticality-radiation alarm system functioned as designed, the evacuation of personnel from the building was prompt and orderly, and the excursion was terminated expeditiously by a negative coefficient of reactivity and was prevented from recurring by the action of the safety devices. The fission yield was 1.1×10^{16}. Gamma-ray sensitive personnel dosimeters read immediately following the excursion showed no direct exposure greater than 5 mr to any person present. There was no property damage or loss of fissile materials. An estimated 100 cm^3 of solution (15 g of U) were spilled when a rubber-stoppered connection immediately above the sphere was dislocated.

The purpose of the particular experiment in progress was to establish the critical concentration of a sphere of the solution of uranyl nitrate surrounded by a thick water reflector. In the course of approaching criticality by incremental additions of solution, a small volume of air was observed entrapped in a flexible transparent tube. Supercriticality occurred during an attempt, by remote manipulation of liquid levels, to remove the air.

2. Fires

Year	Reporting AEC Field Office	
1959	OR	Multiple circuit breaker failure led to severe electrical fire. Property damage $86,000. No exposures.
	LAR	Electrical fire due to severe arcing on the lineside of heater breakers. Property damage $30,000. No exposures.
1960	HA	Fire and explosion in pyrophoric metal contents of a chemical dissolver, off-gas filter, and related process equipment. Contamination spread to cell, canyon and crane. Cause(s) of the accident not established. Property damage $250,443. No exposures.
1962	OR	Fire occurred in ventilation system, probable cause electrical spark. Property damage $24,700. No exposures.
	HA	Air ventilation equipment failure. Property damage $10,000. No exposures.
	OR	Explosion and fire in cell. Property damage $2,900,000. No exposures.
1963	OR	Fire (definite cause undetermined) originated in building exhaust system and was confined to laboratory hoods and exhause system; smoke damaged building. Property damage $43,400. No exposures.
	RL	Fire (definite cause undetermined) in plutonium purification facility. Pu contamination in immediate area of fire. Firemen received slight skin contamination, readily removed. Costs related directly to fire $85,400; decontamination costs $251,300; overhead related to direct losses $60,300.

2. Fires (Continued)

Year	Reporting AEC Field Office	
1964	SR	Fire (definite cause undetermined) occurred around an anion exchange column in hot canyon. Fire caused airborne contamination to crane used for remote maintenance. Water to quench fire damaged electric motors. Property damage $21,000.
1965	RL	During an aluminum jacket dissolution in a dissolver, an exothermic reaction involving ammonia and/or hydrogen occurred with an electric heater, through which these gases were accidentally vented. The reaction, which continued for 3 hours, totally destroyed the heater. Property damage $7,200. No exposures.
1966	SR	A fire occurred when a drying oven overheated. Faulty loading blocked the thermostat sensing element, causing it to indicate erroneously low temperature and call for additional heat. Property damage ($6,000) was confined to the room of origin. No exposures.
	OR	A fire, of undetermined cause, occurred in a laboratory. It was confined to one hood and a section of ductwork because of the successful operation of a sprinkler head, a fire damper in the exhaust system and other fire protection controls. Property damage $5,500. No exposures.
1968	RL	An electrical short circuit and the resulting power arc in the main electrical switchgear damaged two breakers extensively, when the lights went out and the building ventilation stopped. Emergency actions were taken to preclude any contamination spread. Operations were curtailed for two and one-half days while repairs were being made. Property damage $34,000. No exposures.

2. Fires (Continued)

Case History No. - Manufacturing Chemists Association	
41	"Static Spark Flashes "Empty" Styrene Drum"
129	Fire Due to Static Spark - Benzene
141	Toluene Vapors - Flash Fire
150	Escape of Vapor From Condenser
255	Ignition of Solvent Vapors - Employee Burned
341	Solvent Fire
348	Electric Mixer Fires Solvent
612	Waste Solvent Fire
643	Flash Fire in Exhaust Duct
699	Toluene - Static Fire
701	"Boil Over" - Flash Fire
1025	Flammable Solvents - Electric Motor
1217	Kerosene Vapor Flash - Synthesis Kettle
1234	Zirconium Fines Flash Fire
1966	"Static" Ignition of Flammable Solvent
1970	Solvent-Vapor Flash Fire

3. Explosions

Year	Reporting AEC Field Office	
1959	SR	Gasket on head of secondary condenser in unit failed. Relief valve vented open due to overpressure. 4-5T of H_2S gas released to atmosphere. $7,000 property damage. No exposures.
	HA	Plutonium glovebox explosion. $9,500. property damage. No exposures.
	HA	Autoclave explosion. Property damage $4,000. No exposures.
	AL	Drybox explosion. Property damage $1,933. No exposures.
	OR	Chemical explosion in innercycle evaporator. Property damage $350,000. No exposures.
	OR	An explosion occurred in a digester. Property damage $10,000+. No exposures.
1960	OR	Hydrogen gas explosion occurred in gas furnace enclosure in metal plant. Property damage $5,000. One employee suffered serious injuries.
	OR	Explosion occurred in a uranium sintering furnace located in a foundry. Major structural damage to furnace and buildings. Property damage $20,000. No exposures.
	AL	The accidental discharge of radioactive material into a room occurred as a result of pressure buildup in a drybox. This was due to an inlet solenoid being locked in the open position and a venting solenoid being closed due to a malfunction. The pressure built up to a point that one of the drybox gloves blew out, thereby releasing radioactive particulate material into the room. Property damage $31,360. Eleven persons received minor exposures.

3. Explosions (Continued)

Year	Reporting AEC Field Office	
1961	AL	Pressure buildup in closed caustic scrubber system forced airborne radioactive material into room. Property damage $4,016. One employee received 71 rem exposure to bone. Area contaminated.
1963	ID	Low-level spread of plutonium contamination from glovebox. Property damage $25,451. No exposures.
1964	AL	Chemical explosion in metal hood when methanol vapors reached flashpoint. Two sets of gloves were shredded by the explosion. Contamination spread in operating area. Property damage $34,922. Three employees received slight contamination.
1965	RL	An explosion occurred in a boiler during an attempt to relight the oil-fired burner with a kerosene torch after the automatic re-ignition system failed to function. Property damage $75,000. No exposures.
	RL	Plutonium contamination spread following an explosion and fire occurring in a glovebox when cleaning fluid ignited. Ten employees left the building immediately. Prompt showering easily removed all skin contamination. One employee received 10% of a maximum permissible body burden (bone) of Pu-239 by inhalation. Contamination did not spread outside the building. About 90% of the cost ($76,800) was incurred for decontamination.

3. Explosions (Continued)

Year	Reporting AEC Field Office	
1965	AL	An explosion and fire occurred when acetone fumes from a "cocoon" used in a glovebox paint-stripping operation, contacted a hot muffle furnace in another part of the glovebox line. Plutonium contamination spread to adjacent rooms and the second floor. 12 employees required skin decontamination; none received internal radiation exposures. Property damage costs ($23,253) was for decontamination of facilities.
	CH	An explosion resulting from the ignition of a hydrogen-air mixture, the hydrogen apparently evolved from nickel-iron batteries, occurred in the equipment airlock joining a reactor building and a fuel cycle facility. No radioactive material was involved. Property damage $22,600.
	AL	An undetermined small quantity of Pu-238 was released when a double-contained vessel, nearly full of drybox seived material exploded, dispersing a quantity of the waste material into the laboratory. Property damage $19,100. No exposures.

3. Fires (Continued)

Case History No. - Manufacturing Chemists Association

103	Nitrating Operation Explosion
116	Storage Battery Explosion
128	Nitrogen Peroxide - Cyclohexane Mix Explosion
131	Nitric Acid - Waste disposal Explosion
163	Reaction in Solvent Recovery Tank
223	Laboratory Explosion - No Injuries
258	Explosion - Ignited by Vacuum Cleaner
347	Hydrogen dessicator - Drainage Trench Explosion
569	Runaway Nitration Reaction
576	Hydrogen Compressor - Explosion
611	Oil Vapor Explosion
678	Explosion in Nitrobenzene Recovery Kettle
679	Unsafe "Fail Safe" Flame Safety Device
703	Explosion in Vent Stack - Static Generation
859	Spilled Four Gallons Solvent on Laboratory Floor - Fire
880	Chemical Fire - Azido Compound
976	Silver Complex Detonation
987	Explosion and Fire - Lead Azide
988	Tank Explosion
1048	Explosion - Silver Oxide

3. Explosions (Continued)

Case History No. - Manufacturing Chemists Association

1068	Gas Explosion - lighting Burner
1097	Explosion - Hydrogen gas vent stack
1105	Americium Solution Shipping Container Explosion
1188	Hazardous Solvent Causes Explosion in a Plutonium Fuels Laboratory Glovebox (Furnace)
1310	Flammable Vapor Explosion - Slurry Mix Tank
1311	Nitration Explosion - Organic Intermediate Mixed with Nitric and Sulfuric Acids
1496	Drums burst from internal pressure - Accumulation of Hydrogen within
1499	Tank Rupture - Organic Material Nitrated with Nitric Acid
1554	Ammoniacal Silver Nitrate Explosion
1733	Laboratory Explosion - Silver Nitrate, Ammonium Hydroxide, Alcohol Silvering Solution
1957	Disposal of Deposits of Metal Azides
1958	Mix Tank Explosion

4. Fuel Receiving & Storage

Year	Reporting AEC Field Office	
1960	SR	During shipment of irradiated fuel elements, 30 to 40 gallons of contaminated water leaked from the cask. $24,000 cost due to decontamination of area. No exposures.
1964	CH	Broken valve on autoclave, housed in lead shipping cask, allowed contaminated water to seep out of cask during transit, contaminating container and truck floor. No property damage.
1967	RL	A diesel locomotive collided with a cask car during coupling operations, due to the inattention of the diesel's engineer. The cost of $5,124 was for repairing the locomotive; the cask car was not damaged. No exposures.

5. Waste and Product Storage

Year	Reporting AEC Field Office	
1959	SR	Leaking compression fitting. $20,000 damage. No exposures.
	SR	Solution leaked from the loosened flange during maintenance work on a waste evaporator in hot canyon, vaporized and contaminated crane. Property damage $129,324. No exposures.
	SR	Loose contaminated particles on the lid of a waste burial box were scattered by the wind, contaminating the ground, locomotive and spacer car. Property damage $5,200. No exposures.
1960	SR	Contaminated cooling water discharged from canyon onto floor. Property damage $250,000 (due to decontamination). No overexposures.
1961	HA	Uranyl nitrate (1355 lbs. of depleted uranium) lost to ground when tank trailer was overfilled due to misunderstanding between regular operators and their lunch relief. Property damage $13,000. No exposures.
	HA	Approximately 1,089 pounds of depleted uranium lost to chemical sewer in plant. Property damage $9,000. No exposures.
1963	AL	Leak in line carrying high-level plutonium solution caused contamination of building and equipment. Property damage $8,364. No exposures.
	AL	Spill of contaminated nitric acid solution. Property damage $5,662. No exposures.

5. Waste and Product Storage (Continued)

Year	Reporting AEC Field Office	
1964	CH	Clean water, being used to test two new waste tanks, was contaminated by condensation from contaminated vent line connected to one tank. Water subsequently drained onto asphalt surface, contaminating it and drainage ditch. Property damage $6,075. No exposures.
	ID	During steam flushing to remove radioactive contamination from pipelines to permit tie-in to new lines, leak developed in hose coupling. Contaminated fluid and steam issuing from leak were rapidly dispersed by high wind over approximately 10 acres. Majority of $12,884 cost due to cleanup.
	SR	Leaking nitric acid corroded canyon cell equipment beyond repair. Property damage $6,000. No exposures.
1965	AL	While attempting to activate a product transfer line, contaminated acid solution was sprayed out of a flanged union that had not been tightened. Three contaminated employees were readily decontaminated. Property damage $7,557 for decontamination and for replacing contaminated equipment.
	SR	Process water (2400 lbs) was lost when it leaked through an unseated sleeve. Property damage $33,600. No exposures.
	SR	Process water (700 lbs) was lost when it leaked through an unseated sleeve. Property damage $9,800. No exposures.
	SR	A cooling coil in a vessel developed a leak and allowed contaminated solution from a tank to enter the cooling water system when the coil was pressurized. The cost for cleaning the system and associated work was $19,500. No exposures.

5. Waste and Product Storage (Continued)

Year	Reporting AEC Field Office	
1966	RL	An estimated 420 lbs. of uranium solution were lost to radioactive waste through a milling tank overflow, caused by the failure of a normally closed supply line valve. Property damage $7,200. No exposures.
	RL	During the repair of an air circulation valve, approximately 10 gallons of high-level radioactive waste solution were spilled onto the floor. Three employees, wearing protective clothing, were sprayed with droplets of the solution, but were readily decontaminated. Property damage $19,746.
	RL	Less than 5 grams of concentrated plutonium nitrate solution spilled on the elevator floor when a product receiver assembly overturned and the lid of the inner container came off. Cost for decontamination was $13,443. No exposures.
1967	AL	Abandoned storage vessels inside a stainless steel glovebox were being flushed with 7-9N nitric acid to recover plutonium nitrate. During this operation, the air monitor alarmed, and the odor of nitric oxide fumes was detected. Shortly thereafter, a puddle of dark liquid (plutonium nitrate solution) was seen on the floor under the glovebox. The solution had leaked from one of the storage vessels into the glovebox well and thence onto the floor of the room. 3 contaminated personnel were readily decontaminated by showering; the cost for decontamination of the storage vessel area was $16,465.

5. Waste and Product Storage (Continued)

Year	Reporting AEC Field Office	
1967	SR	Radioactive liquid waste was stored in an underground tank. The pipeline for the concentrate entered the storage tank through a shielded riser, extending from the top of the tank to approximately one foot above ground. Access plugs were sealed with mastic compounds. The inlet pipe entered the riser horizontally below ground and terminated with a valve near the center of the riser. When the riser became plugged with concentrate crystals below the inlet line, the liquid flow reversed and forced its way through the access plugs. Approximately 13 curies of radioactive liquid waste, primarily cesium, were released to plant streams but sampling showed that radioactivity concentration standards were not exceeded in streams beyond the plant boundary. The cost for decontamination of ground in the vicinity of the tank was $49,179. No exposures.
1969	SR	Approximately 20 millicuries of airborne radioactive contamination (mostly curium-244) were released via an exhaust stack and were spread by a northeasterly wind across the roof of a building and along a line leading from the main entrance of the building to a parking lot. The level of radioactivity on the roof was 4×10^7 d/m/100cm^2 and on vehicles inside the area fence 1.5×10^4 d/m/100cm^2. The highest level of activity outside the area was approximately 5000 d/m/100cm^2. All activity was contained within the plant's boundary. The cost for decontamination was $37,506. No radiation exposures.
	SR	Acidic waste solution (approximately 8200 pounds), which was being processed for neptunium-237 and plutonium-238 recovery, was lost when inadvertently transferred to an underground waste system due to a leaking valve in the stream supply to a transfer jet. Property damage $32,000. No exposures.

5. Waste and Product Storage (Continued)

Year	Reporting AEC Field Office	
1970	SR	A solution containing 20 grams of curium-244 and americium-243 was transferred by mistake to the waste system. Property damage $124,523. No exposures.
	RL	Minor cerium-cesium contamination resulted from a routine change of a stack filter in the 300 area. Over 200 employee's shoes were checked and none were found to be contaminated. Walkways and roadways were washed down; no radioactivity was found in surveys beyond the 300 area. No property damage; no exposures.

5. Waste and Product Storage (Continued)

Case History No. - Manufacturing Chemists Association	
254	Pressure Build-Up in Pfandler Kettle Operation
298	Collapse of 20,000 gallon S.S. Tank
1088	Implosion in Still During Cleaning Operation
1498	Gross Leakage of Plutonium Nitrate Solution from Stored Polyethylene Bottles
1716	Dry Radioactive Waste Unloading Incident - Localized Radioactive Dust Escape

6. Natural Phenomena

Year	Reporting AEC Field Office	
1959	OR	Lightning damaged transformer. Property damage $6,500. No exposures.
	SR	Lightning damaged two 750 KVA transformers. Property damage $13,750. No exposures.
	LAR	Wind damage to aluminum side wall of building. Property damage $7,500. No exposures.
	SR	Hurricane damaged water dam. Property damage $50,000. No exposures.
1960	SR	During an electrical storm, lightning struck two 200-hp pump motors in an out-of-door pump pit. $6,000 cost for rewinding of burned out motors. No exposures.
	OR	During violent storm, severe power system disturbance caused oil circuit breaker failure. Property damage $18,132. No exposures.
	AL	High-velocity winds caused circuit breaker failure in substation, resulting in fire readily controlled by fire extinguisher. Property damage $8,200. No exposures.
1963	PNR	Severe winds during electrical storm damaged roofs, stacks, ventilation ducts, trees and fences. Property damage $9,400. No exposures.
1965	CH	Four transformers were damaged by lightning. Property damage $35,000. No exposures.
	CH	Lightning caused the destruction of a breaker and the burning of a cubicle. Property damage $8,000. No exposures.
	AL	Repeated lightning strikes damaged transformers. Property damage $5,400.

6. Natural Phenomena

Year	Reporting AEC Field Office	
1966	AL	Roofing destroyed by high winds. Property damage $47,000. No exposures.
1967	AL	A severe wind and hailstorm, with winds in the range of 80 to 100 miles per hour and hailstones the size of oranges, caused extensive roof and other structural damage to numerous buildings, disrupted utilities, demolished a warehouse wall, leveled security fencing and caused severe vehicle damage. Property damage $1,872,000. No exposures.
1970	SR	Water supply lines, drainlines and traps, water-jacketed equipment, heating and cooling coils, instruments, gages, and fire sprinkler lines froze in numerous plant locations during a period of extremely low temperatures, unusual and unexpected in the area. Property damage $38,200. No exposures.